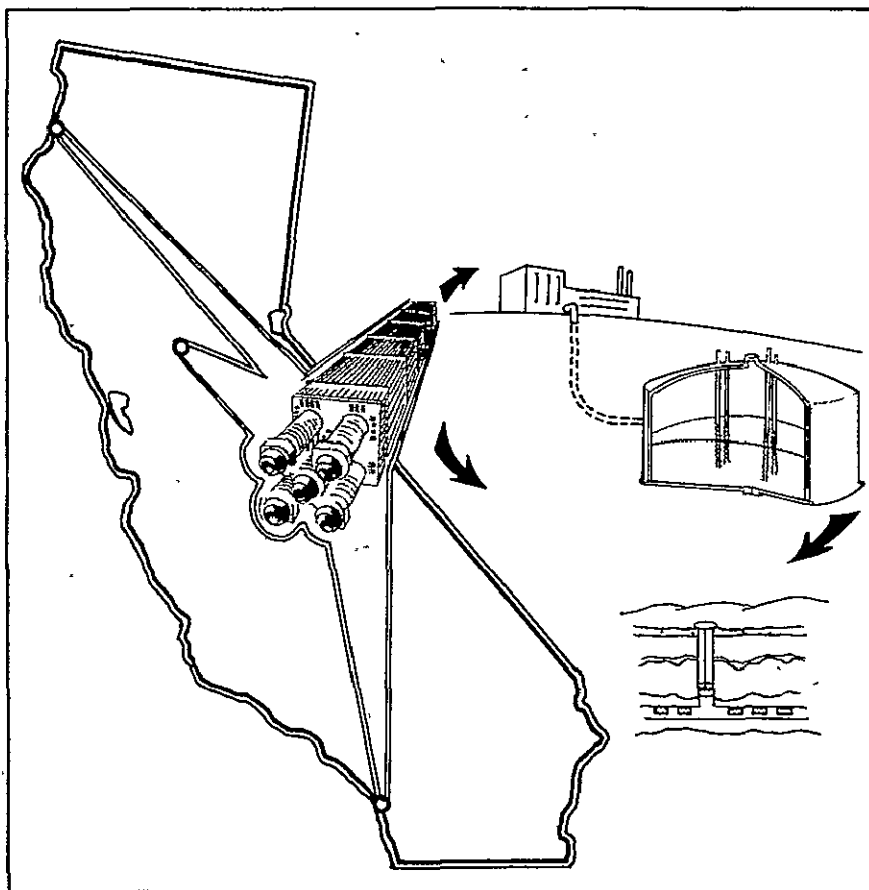


An Analysis of the Technical Status of High Level Radioactive Waste and Spent Fuel Management Systems

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Prepared for

California Energy Resources Conservation
and Development Commission

by

Jet Propulsion Laboratory
California Institute of Technology
Pasadena, California 91103



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An Analysis of the Technical Status of High Level Radioactive Waste and Spent Fuel Management Systems

Dr. T. English, Project Manager
Dr. C. Miller, Deputy Project Manager
Sir E. Bullard
Dr. R. Campbell
Mr. A. Chockie
Mr. E. Divita
Mr. C. Douthitt
Mr. E. Edelson
Prof. L. Lees

December 1, 1977

Prepared for

California Energy Resources Conservation
and Development Commission

by

Jet Propulsion Laboratory
California Institute of Technology
Pasadena, California 91103

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FOREWORD

This report summarizes a portion of the results of a research project conducted by the Nuclear Waste Management Project of the Jet Propulsion Laboratory for the California Energy Resources Conservation and Development Commission. The purpose of this report was to examine the technical status of the United States' principal programs for the management and disposal of high-level radioactive waste. Two waste forms were included in the analysis:

- Solidified High-level Waste
- Spent Fuel Rods

This study was performed by a group drawn from the Jet Propulsion Laboratory, the campus of the California Institute of Technology, and the Scripps Institution of Oceanography. The duration of the study was approximately seven months, and involved three man-years of effort.

The work was funded under California ERCDC Standard Agreement No. 154-002. The Project Coordinator was Dr. James A. Walker, Deputy Executive Director, California Energy Resources Conservation and Development Commission.

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We are especially grateful to David Dawson and James Boyden of the General Electric Company (San Jose, CA) and J. M. McElroy, A. L. Platt, and Wayne Ross of the Battelle Northwest Laboratory (Richland, WA) for their time and effort in providing us with information and for technically reviewing segments of our draft report. We are grateful for the assistance of Lawrence Baker & Edward Koprowski in preparing the Glossary -- Appendix B. In addition, we are thankful for the patience and skill of our secretary, Jeanne Wadsworth.

DEFINITION OF ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safety
ACVC	Air-cooled Vault Concept
A-E	Architect-Engineer
AEC	Atomic Energy Commission
AFR	Away-from-reactor (spent fuel storage facility)
AGNS	Allied-General Nuclear Services
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
APS	American Physical Society
BeV	Billion Electron Volts
BNFP	Barnwell Nuclear Fuel Plant
BNWL	Battelle Northwest Laboratories
BWR	Boiling Water Reactor
CANDU	Canadian Deuterium Reactor
CEDM	Control Element Drive Mechanism
CEQ	Council on Environmental Quality
CERDC	California Energy Resources Conservation and Development Commission
CFR	Code of Federal Regulations
CHIC	Chicago
CIT	California Institute of Technology
CRWM	Committee on Waste Management (NAS National Research Council)
D ₂ O	Heavy water
DGEIS	draft generic environmental impact statement
EIS	environmental impact statement

EPA	Environmental Protection Agency
EPA W	Environmental Protection Agency Workshop
ERCDC	Energy Resources Conservation and Development Commission (California)
ERDA	Energy Research and Development Administration
EPRI	Electric Power Research Institute
FEIS	Final Environmental Impact Statement
FRP	Fuel Reprocessing Plant
FSAR	Final Safety Analysis Report
FSER	Final Safety Evaluation Report
GAO	General Accounting Office
GE	General Electric
GESMO	General Environmental Impact Statement for Mixed Oxide Fuels
GSD	Genetically Significant Dose
GWe	Gigawatt electrical
HEPA	High-efficiency Particulate Air filter
HLLW	High-level Liquid Waste
HLW	High-level Waste
HTGR	High-temperature Gas-cooled Reactor
INEL	Idaho National Engineering Laboratory
JPL	Jet Propulsion Laboratory
LLL	Lawrence Livermore Laboratories
LLW	Low-level Waste
LMFBR	liquid-metal fast breeder reactor
LWR	light-water reactor
MeV	Million Electron Volts
MFRP	Midwest Fuel Recovery Plant
MIT	Massachusetts Institute of Technology

MOX	Mixed oxide (reactor fuel)
MPC	Maximum permissible concentration
MPCA	Maximum permissible concentration in air
MPCW	Maximum permissible concentration in water
mrem	Millirem
MT	Metric Ton
MTU	Metric Ton of Uranium fuel
MWd	Megawatt day
MWe	Megawatt electrical
MWt	Megawatt thermal
NASA	National Aeronautics and Space Administration
NFS	Nuclear Fuel Services
NRC	National Regulatory Commission
NRDC	National Resources Defense Council
NWTS	National Waste Terminal Storage
OMB	Office of Management and Budget
OSTP	Office of Science and Technology Policy (Executive Office of the President of the United States)
OTA	Office of Technology Assessment
OWI	Office of Waste Isolation
PG&E	Pacific Gas and Electric Company
PNL	Pacific Northwest Laboratories
PWR	Pressurized Water Reactor
Quad	10 ¹⁵ Btu
r	Roentgen
rad	Radiation Absorbed Dose
RBE	Relative Biological Effectiveness
rem	Roentgen Equivalent Man

RSSF	Retrievable Surface Storage Facility
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SFA	Spent Fuel Assembly
SIO	Scripps Institution of Oceanography
SNAP	Systems for Nuclear Auxiliary Power
SNM	special nuclear material
SSCC	sealed storage cask concept
SURFF	spent unprocessed fuel facility
SWU	separative work unit
TAD	Technical Alternatives Document
TBP	tributylphosphate
U of A	University of Arizona
UCSD	University of California at San Diego

ABSTRACT

Analyses were made of the technical status of the "old U.S. mainline program" for high-level radioactive nuclear waste management, and the newly-developing program for disposal of unprocessed spent fuel. The method of long-term containment for both of these waste forms is considered to be deep geologic isolation in bedded salt. This analysis was designed to assist the members and staff of the California Energy Resources Conservation and Development Commission in carrying out the mandate of California Assembly Bill AB 2822. Each major component of both waste management systems is analyzed in terms of its Scientific Feasibility, Technical Achievability and Engineering Achievability. The resulting matrix leads to a systematic identification of major unresolved technical or scientific questions and/or "gaps" in these programs.

CONTENTS

1	SUMMARY -----	1-1
1.1	"OLD MAINLINE HLW" PROGRAM -----	1-1
1.2	DISPOSITION OF UNREPROCESSED SPENT FUEL -----	1-2
1.3	GEOLOGIC DISPOSAL OF ENCAPSULATED HIGH-LEVEL WASTE AND UNREPROCESSED SPENT FUEL IN BEDDED SALT -----	1-3
2	INTRODUCTION -----	2-1
3	METHODOLOGY FOR ANALYZING TECHNOLOGY STATUS -----	3-1
3.1	TERMINOLOGY -----	3-1
3.2	STAGES OF TECHNOLOGY DEVELOPMENT -----	3-3
3.3	SYSTEM REQUIREMENTS -----	3-6
3.4	INFORMATION SOURCES -----	3-6
3.5	GENERAL ANALYTIC APPROACH -----	3-8
3.5.1	Public Policy Constraints -----	3-14
3.5.2	Identification and Analysis of Issues -----	3-14
4	OVERVIEW OF HIGH-LEVEL WASTE AND AND SPENT FUEL SYSTEMS -----	4-1
4.1	ORIGIN AND NATURE OF SPENT FUEL -----	4-1
4.2	ORIGIN AND NATURE OF HIGH LEVEL WASTE -----	4-4
4.3	RADIATION AND HEAT PROPERTIES -----	4-4
4.4	PROJECTIONS OF SPENT FUEL FROM U.S. COMMERCIAL LIGHT-WATER REACTORS -----	4-6
4.4.1	Nuclear Energy Capacity and Spent Fuel Discharged -----	4-7
4.4.2	Transportation -----	4-7
4.5	GENERATION OF HIGH-LEVEL WASTE -----	4-9
4.6	SYSTEMS FOR STORAGE AND DISPOSAL OF SPENT FUEL AND SOLIDIFIED HIGH LEVEL WASTE -----	4-9

5	ASSESSMENT OF STATUS OF "OLD MAINLINE" PROGRAM FOR DISPOSAL OF SOLIDIFIED HIGH-LEVEL WASTE -----	5-1
5.1	LIQUID CONTAINMENT OF HIGH-LEVEL WASTE -----	5-1
5.1.1	Process Description -----	5-1
5.1.2	History of Storage of Liquid Waste -----	5-2
5.1.3	Liquid-Acidic High-Level Waste Containment Program Status and Plans -----	5-7
5.1.4	Issue Related to Technology Status of Liquid Containment -----	5-8
5.2	TREATMENT AND PACKAGING -----	5-8
5.2.1	Solidification of Waste by Spray Calcination -----	5-8
5.2.1.1	Process Description -----	5-8
5.2.1.2	Development History -----	5-10
5.2.1.3	Program Status and Plans -----	5-10
5.2.1.4	Issues Related to Technology Status of Spray Calcination -----	5-12
5.2.1.4.1	Scientific Feasibility -----	5-12
5.2.1.4.2	Technical Feasibility -----	5-12
5.2.1.4.3	Engineering Achievability -----	5-13
5.2.2	Immobilization of Solidified Waste by Vitrification -----	5-17
5.2.2.1	Process Description -----	5-17
5.2.2.2	Development History -----	5-19
5.2.2.3	Program Status and Plans -----	5-20
5.2.2.4	Issues Related to Technology Status of In-Can Vitrification -----	5-20
5.2.2.4.1	Scientific Feasibility -----	5-20
5.2.2.4.2	Technical Feasibility -----	5-26
5.2.2.4.3	Engineering Achievability -----	5-29
5.2.3	Encapsulation -----	5-31

5.2.3.1	Process Description -----	5-31
5.2.3.2	Development History -----	5-31
5.2.3.3	Program Status and Plans -----	5-31
5.2.3.4	Issues Related to Technology Status of Encapsulation -----	5-33
5.2.3.4.1	Scientific Feasibility -----	5-33
5.2.3.4.2	Technical Feasibility -----	5-34
5.2.3.4.3	Engineering Achievability -----	5-34
5.3	DEEP GEOLOGIC ISOLATION IN BEDDED SALT -----	5-35
5.3.1	Reference Repository Description -----	5-35
5.3.2	Development History -----	5-37
5.3.3	Program Status and Plans -----	5-38
5.3.4	Issues Related to Technology Status of Bedded Salt Isolation -----	5-42
5.3.4.1	Scientific and Technical Feasibility -----	5-42
5.3.4.1.1	Geology and Site Criteria Development -----	5-43
5.3.4.1.2	Salt Mining and Emplacement/Retrievability Technology -----	5-45
5.3.4.1.3	Experience in Storage of Radioactive Material in Salt Deposits -----	5-46
5.3.4.1.4	Salt-Radioactive Waste Interactions -----	5-47
5.3.4.1.5	Release and Transport of Radionuclides to the Biosphere -----	5-49
5.3.4.2	Engineering Achievability -----	5-50
6	STATUS OF AND ISSUES RELEVANT TO COMPONENTS OF THE SPENT FUEL STORAGE AND DISPOSAL SYSTEM -----	6-1
6.1	SPENT FUEL STORAGE AT REACTOR SITES -----	6-1
6.1.1	Refueling Preparations -----	6-1
6.1.2	Removal and Storage of Spent Fuel -----	6-3

6.1.3	Loading and Preparation of Cask for Off-Site Shipping -----	6-5
6.1.4	Development History for Reactor Pool Storage -----	6-6
6.1.5	Status and Plans -----	6-7
6.1.6	Issues - Engineering Achievability -----	6-8
6.1.7	Design Approach, Constraints, Regulations -----	6-8
6.2	INDEPENDENT SPENT FUEL POOL STORAGE -----	6-9
6.2.1	Process Description -----	6-9
6.2.2	Development History -----	6-16
6.2.3	Status and Plans -----	6-22
6.2.4	Issues--Engineering Achievability -----	6-24
6.2.4.1	Direct Radiation -----	6-25
6.2.4.2	Liquid Release to Environs -----	6-31
6.2.4.3	Airborne Release to the Environs -----	6-31
6.3	ENGINEERED INTERIM SURFACE OR SHALLOW SUBSURFACE STORAGE -----	6-33
6.3.1	Process Description -----	6-34
6.3.1.1	Packaging -----	6-36
6.3.1.2	The Storage Facility -----	6-37
6.3.2	Development History -----	6-42
6.3.3	Program Status and Plan -----	6-45
6.3.4	Issues -----	6-46
6.3.4.1	Scientific Feasibility - The Effects of Reactor Environment on the Fuel Rods' Material Integrity -----	6-46
6.3.4.2	Technical Feasibility - Related Technical Issues Addressed in the NAS Review of the RSSF -----	6-47
6.4	DEEP GEOLOGIC DISPOSAL OF UNREPROCESSED SPENT FUEL IN BEDDED SALT -----	6-48
6.4.1	Description of Repository and Packaging Facility Concepts -----	6-48

6.4.2	Developmental History and Program Status and Plans ----	6-49
6.4.3	Technical Aspects of Spent Fuel Repository Design and Packaging; Major Design Parameters -----	6-51
6.4.3.1	Radioactivity and Decay Heat of Spent Fuel -----	6-51
6.4.3.2	Packaging, Canister Spacing and Temperature Effects on Bedded Salt -----	6-53
6.4.3.3	Long-Range Considerations -----	6-59
7	STATUS AND ISSUES OF SYSTEM REQUIREMENTS -----	7-1
7.1	INTRODUCTION -----	7-1
7.2	PROCESS DESCRIPTION -----	7-1
7.2.1	Transportation -----	7-2
7.2.2	Facilities Required -----	7-3
7.2.3	R&D Program -----	7-4
7.2.4	Cost and Ownership -----	7-4
7.2.5	Institutional Aspects -----	7-5
7.3	FLOWCHARTS OF U.S. MAINLINE PROGRAM FOR COMMERCIAL HIGH-LEVEL NUCLEAR WASTES -----	7-6
7.3.1	Timing of the Development of Required Facilities-----	7-10
8	CONCLUSIONS -----	8-1
8.1	"OLD MAINLINE HLW" PROGRAM -----	8-1
8.1.1	Interim Liquid Containment of High-Level Waste -----	8-1
8.1.2	Solidification of the Liquid in a Spray Calciner -----	8-1
8.1.3	Immobilization of Solidified Waste by Vitrification ---	8-2
8.1.4	Encapsulation -----	8-3
8.2	DISPOSITION OF UNREPROCESSED SPENT FUEL -----	8-4
8.2.1	Spent Fuel Storage at Reactor Sites -----	8-4
8.2.2	Independent Off-Site or On-Site Spent Fuel Storage ----	8-5
8.2.3	Engineering Interim Surface or Shallow Sub-Surface Storage (SURFF Storage) -----	8-7

8.2.4	Packaging of Unreprocessed Spent Fuel for Disposal -----	8-8
8.3	GEOLOGICAL DISPOSAL OF ENCAPSULATED HIGH-LEVEL WASTE AND UNREPROCESSED SPENT FUEL; DEEP GEOLOGIC ISOLATION IN BEDDED SALT -----	8-8
8.3.1	General Conclusions -----	8-8
8.3.2	Technical Aspects of Repository Design: Packaging, Canister Spacing and Temperature Effects on Bedded Salt -----	8-8
8.3.3	Long-Term Repository Integrity -----	8-9
8.4	Matrix Summary -----	8-12
8.5	SYSTEM REQUIREMENTS -----	8-12

APPENDICES

A	ORIGIN, NATURE AND DISPOSAL OF HIGH-LEVEL WASTE -----	A-1
B	GLOSSARY OF SPECIAL TERMINOLOGY -----	B-1
C	SUMMARY OF EXTERNAL REVIEW COMMENTS ON DRAFT REPORT -----	C-1
D	CALIFORNIA ASSEMBLY BILL NO 2822 -----	D-1

REFERENCES

SECTION 2	-----	2-5
SECTION 3	-----	3-16
SECTION 4	-----	4-20
SECTION 5	-----	5-50
SECTION 6	-----	6-63
SECTION 7	-----	7-11
APPENDIX A	-----	A-23

LIST OF FIGURES

2-1	Major U.S. High Level Waste Disposal Programs -----	2-2
3-1	Visits of JPL/CIT/SIO Nuclear Waste Management Team ---	3-10
3-2	General Analytic Approach -----	3-11
3-3	Relationship Between Public Policy and Stages of Technology Development Showing Guidelines -----	3-12
3-4	Blank Matrix Summary of the Technical Status of Com- ponents in the U.S. Mainline High-Levels Waste Disposal System -----	3-13
4-1	Thermal Power Released by the Radioactivity of High-Level Waste from Reprocessing One Metric Ton of Irradiated LWR Fuel -----	4-2
4-1A	Radioactivity for PWR Spent Fuel -----	4-3
4-2	Comparison of Nuclear Power Growth Estimates -----	4-8
4-3	Accumulation of LWR Fuel Assemblies -----	4-10
4-4	Cumulative MTU Discharged -----	4-11
4-5	Annual Shipment of Spent Fuel -----	4-13
4-6	Cumulative High-Level Waste -----	4-16
4-7	Systems of Interest -----	4-18
5-1	Typical Storage Tank for Nonboiling Waste -----	5-5
5-2	Typical Control and Safety Features for High-Level Boiling Radioactive Waste Storage -----	5-5
5-3	Double-Wall Boiling Waste Tank -----	5-6
5-4	High-Level Liquid-Waste Process Flow Diagram -----	5-9
5-5	Diagram of Spray Calciner -----	5-11
5-6	Remote Design Features of the Spray Calciner -----	5-14
5-7	Cell Arrangement of Vitrification Equipment, Plan View -----	5-15
5-8	Cell Arrangement of Vitrification Equipment, Elevation View -----	5-16

5-9	In-Can Melter -----	5-18
5-10	Leachability of ^{244}Cm -Doped Zinc Borosilicate Waste Glass -----	5-22
5-11	Effect of Cooling Rate on the Generation of Increased Surface Area in a 6 inch Diameter Glass Casting -----	5-24
5-12	Range of Composition of HLW Glasses -----	5-28
5-13	Multizone Resistance Heated-Air Cooled Melter Furnace -----	5-30
5-14	Typical Post-Fill Treatment of Solidified HLLW Canisters -----	5-32
5-15	Simplified Cutaway of Repository -----	5-36
5-16	Rock Salt Deposits in the United States -----	5-39
5-17	Geologic Terminal Storage, Repository No. 1 Design and Construction Schedule -----	5-40
5-18	WIPP Milestone Chart -----	5-41
6-1	Arrangements of Major Components of the Fuel Handling System -----	6-2
6-2	Fuel Transfer System -----	6-4
6-3	GE IF-300 Irradiated Fuel Shipping Cask -----	6-11
6-4	IF-300 Cask in Receiving Area -----	6-12
6-5	Cask Decontamination Area -----	6-13
6-6	IF-300 Cask on Cask Shelf -----	6-14
6-7	Cask Unnloading Basin Deep Pit -----	6-15
6-8	Fuel Grappling Operations -----	6-17
6-9	Fuel Removal from Cask -----	6-18
6-10	Fuel Basket Loading -----	6-19
6-11	Fuel Basket Transfer -----	6-20
6-12	Fuel Movement in Storage Basins -----	6-21
6-13	Fuel Storage Basin Areas -----	6-23
6-14	Event Diagram for Postulated Accidents -----	6-26

6-14A	Stainless Steel Basin Liners -----	6-28
6-14B	Stainless Steel Pool Liner -----	6-30
6-15	Concrete Surface Silo, Convection Cooled Concept -----	6-40
6-16	RSSF Air-Cooled Vault Concept ¹⁰ -----	6-41
6-17	Typical Concrete Tile Hole for Surface Storage of CANDU Fuel -----	6-43
6-18	Thermal Power Per Metric Ton of Uranium Versus Time -----	6-52
6-19	Schematic of Isotherms for HLW Disposal Site -----	6-58
7-1	Technology Demonstration and Approval Path -----	7-7
7-2	Timing Chart for High-Level Nuclear Waste Management --	7-11
8-1	Matrix Summary of the Technical Status of Components of the "Old Mainline" High-level Waste Disposal System -----	8-13
8-2	Matrix Summary of the Technical Status of Components of the Newly Developing Waste Disposal System for Unreprocessed Spent Fuel -----	8-14
A-1	PWR Fuel Rod -----	A-3
A-2	PWR Fuel Assembly -----	A-4
A-3	Radioactivity of Re-processed Waste derived from 1 ton of Uranium Fuel -----	A-9
A-4	Hazard Indices for Spent Fuel and Re-processed Waste derived from 1 ton of Fuel -----	A-15
A-5	Underground Storage in Bedded Salt -----	A-19

Tables

3-1	Features of Stages of Technology Development -----	3-5
4-1	Materials in High-level Liquid Waste -----	4-5
4-2	Comparison of Past Forecasts Energy-Electrical Capacity-Nuclear Capacity Year 2000 -----	4-9
4-3	Cumulative MTU Discharged -----	4-12
4-4	Annual Shipments of Spent Fuel -----	4-14

4-5	Generation of Solidified High-Level Waste -----	4-15
5-1	High Level Nuclear Waste Storage Capacity in Cubic Meters, 1971 -----	5-3
5-2	Long-Term Leach Data for the Reference Glass -----	5-22
6-1	Summary Comparison of Spent Fuel Storage Alternatives -----	6-35
6-2	Comparison of Retrievable Surface-Storage Facility Concepts -----	6-38
6-3	Terminal Storage Program -----	6-51
6-4	Design Characteristics for Spent Fuel Disposal -----	6-56
6-5	Parameters of the Geologic Formations -----	6-63
6-6	Plutonium Migration under Hypothesis 2 (Not Sorbed) ---	6-63
6-7	Plutonium -239 Sorbed -----	6-63
A-1	Radioactivity of fission products -----	A-8
A-2	Radioactivity of transuranic elements -----	A-9
A-3	Amounts of a few radioactive isotopes allowed in air and water for unrestricted use -----	A-13

SECTION 1

SUMMARY

This report analyzes the technical status of the "old U.S. mainline program" for high-level radioactive nuclear waste management, plus the newly-developing program for disposal of unprocessed spent fuel. The method of long-term containment for both of these waste forms is considered to be deep geologic isolation in bedded salt. This analysis is designed to assist the members and staff of the California Energy Resources Conservation and Development Commission in carrying out the mandate of AB 2822. Each major component of both waste management systems is analyzed in terms of its Scientific Feasibility, Technical Achievability and Engineering Achievability. The resulting matrix leads to a systematic identification of major unresolved technical or scientific questions and/or "gaps" in these programs, in the context of paragraphs (a) and (c) of Section 25524.2 (AB 2822) of the California Public Resources Code (see Section 2).

To summarize the findings of this analysis, the following conclusions and observations appear to be the most important.

1.1 "OLD MAINLINE HLW" PROGRAM

Interim storage and containment of liquid waste has reached the stage of Engineering Achievability. Since a "hot" pilot plant is planned for 1978, spray calcination and in-can melting prior to vitrification is approaching Engineering Achievability. A level of waste oxides of 23 wt.% has been selected based on considerations of processing factors, leachability, and economic factors. However, the ability of the borosilicate glass to retain the waste's radionuclides is influenced by the response of the glass to the radioactive, thermal, mechanical, and chemical properties of the waste. The individual effects of each waste property have been investigated and detrimental impacts on the glass have been observed. Continued investigations are required in order to fully understand the combined effects that the waste's properties could produce on the glass. The actual amount of devitrification from the combined effects must be determined as well as the magnitude of the other factors which influence the potential leaching rate of the radionuclides from the glass. Further studies are planned to investigate the reference glass as well as non-reference glass forms in greater detail. Hence, the technical achievability of vitrification with commercial high-level waste is in the process of being demonstrated. A "hot" pilot plant incorporating spray calcination, in-can melting and vitrification is planned for 1978. Hence, some components of the vitrification system are approaching engineering achievability.

1.2 DISPOSITION OF UNREPROCESSED SPENT FUEL

For unprocessed spent fuel management, the major components analyzed are (1) Interim storage in existing spent fuel pools at reactor sites, including re-racking to significantly increase storage capacity; (2) Interim storage in independent on-site or off-site spent fuel pools; (3) Interim (20-30 years) dry storage of spent fuel in an engineered surface or shallow sub-surface retrievable facility (SURFF).

By using more accurate criticality codes and fixed "poisons" in the racks, new high-density spent fuel storage racks have been designed and constructed. As a result, re-racking is now feasible to increase the storage capacity of an existing reactor pool from 1 1/3 or 1 2/3 reactor cores up to 4 cores. Allowing for a full-core reserve, this expansion would permit up to 9 years of storage capacity for the storage of spent fuel for a single reactor. As of July 1, 1977, 16 nuclear reactors had obtained NRC approval for re-racking. If NRC approves all 18 pending requests for re-racking, and if intra-utility and inter-utility transfers are permitted were necessary, a net additional storage capacity of 8,200 MTU would be available for future discharge. This is the equivalent of 6 years cumulative discharge for all 63 existing reactors operating at a 70% load factor.

If the decision to reprocess spent fuel is deferred indefinitely, the utilities will be faced with some difficult choices in the next few years: (1) Provide additional independent on-site spent fuel pool storage; (2) Join in a private consortium to construct independent off-site spent fuel pool storage, or contract with potential builders and operators of such facilities (such as Exxon, for example); (3) Declare the spent fuel as "waste" and call upon the Federal government to provide storage. These decisions will be influenced strongly by economic factors such as storage costs and the estimated value of the spent fuel, should processing be approved at some future date. It is conceivable that the Federal government will have to assume responsibility for construction and operation of interim off-site spent fuel storage facilities as insurance against the possibility of unforeseen delays in the 1985 operational schedule for the Federal repository in bedded salt. This decision by the Federal government may also be influenced strongly by the need to provide storage capability for foreign reactors operating on enriched uranium fuel provided by the United States.

In addition to these economic and international political factors, there are several important technical factors to be considered in designing a spent fuel storage facility so as to minimize radiation hazards to the public: (1) release of direct radiation from the irradiated fuel; (2) liquid release to the environs from either an excessive storage basin leak or a low-activity storage vault leak; (3) airborne release from a fuel cladding rupture or from a cooler leak. Solutions to all of these technical problem areas have reached the stage of Engineering Achievability.

1.3 GEOLOGIC DISPOSAL OF ENCAPSULATED HIGH-LEVEL WASTE AND UNREPROCESSED SPENT FUEL IN BEDDED SALT

The experiments conducted in Project Salt Vault demonstrated that the technical and engineering problems associated with (1) The safe handling of radioactive materials in an underground environment; (2) The behavior of the bedded salt in response to the radioactivity and decay heat emitted by the encapsulated waste (or spent fuel); (3) The structural design of the underground repository, are all solvable with currently available technology.

The first major issue is the duration of pre-storage or "aging" before emplacement in the repository. However, our findings show that if the pre-storage time is 10 years, the maximum temperature rise at the canister wall is well below the level at which any significant difficulties are expected because of brine migration or because of salt decrepitation.

A second major issue is the "decay heat" rate per metric ton of uranium. This "decay heat" rate for spent fuel is much larger than for high-level waste in the periods 10³-10⁵ years after discharge from the reactor. In order to match the integrated heat load for HLW over the time interval 10-10⁴ years, the areal power density for the spent fuel in the disposal layer in kw/acre must be about 1/5 that for HLW (20 kw/acre vs 100 kw/acre). The number of canisters required for spent fuel is therefore about 75 per GW(e)-year compared to 15 for HLW.

At the present time long-term repository integrity is an unresolved issue, and is judged to be in the scientific feasibility stage. The weakest links are in the site evaluation and selection processes as follows:

- (1) Specification of hydrogeological criteria necessary to ensure site stability, including seismic criteria
- (2) Understanding, measurement, and modeling of the current ground water regime in any given area of interest
- (3) Extrapolation of that ground water regime up to one million years into the future
- (4) Understanding of geospheric transport mechanisms, and especially the sorption factors for the long-lived radionuclides in the actual geophysical environment.

Experience to date with the sites at Lyons, Kansas; Asse, West Germany; and Carlsbad, New Mexico, show that a number of "scenarios" can be constructed in which a connection between the repository and the biosphere is established. The most important geospheric transport mechanism is the intrusion of flowing ground water. The results of a simplified one-dimensional calculation show that: (1) Without sorption, even an excellent confining geologic formation only introduces a delay in the transport of the radionuclide to the environment; it does not

retain it; (2) With a high degree of sorption even a relatively poor geologic formation retains ^{239}Pu (for example) for a sufficiently long time so that radioactive decay prevents it from reaching the environment. These results emphasize the necessity for a comprehensive program of laboratory and field studies on the interactions of radionuclides with both individual minerals and rocks.

The (OWI) has the responsibility for commissioning and evaluating a set of studies and experiments designed to answer some of these questions. Their studies to date include: (1) Identification of salt formations of interest; (2) Reconnaissance studies; (3) Area studies. This work has not yet progressed to the next levels of detailed confirmation studies and in-situ tests. In addition, ERDA and NRC have not developed design criteria, site suitability criteria, and site selection criteria. Thus extrapolation of existing information to predict Engineering Achievability is very difficult.

At the present time the Scientific Feasibility of deep geologic disposal of encapsulated high-level nuclear waste and unprocessed spent fuel in bedded salt has not yet been proven. There is no scientific reason, however, to anticipate that existing, planned and recommended work will not result in a scientific data base that will permit site selection and design, although the proposed time schedule may be too optimistic. During the 10-20 year period of testing, monitoring, data analysis, and projections, the geologic repository will necessarily be operated in a retrievable mode. This procedure has two important favorable by-products: (1) By storing spent fuel in the repository, the pressure for continual expansion of on-site or independent off-site spent fuel pool capacity is considerably reduced, if not eliminated, and the need for SURFF may also be eliminated; (2) If a decision to proceed with reprocessing should be made during this period, spent fuel could be retrieved and transported to an off-site or to an adjacent fuel reprocessing plant. The solidified high-level waste discharged from the FRP could be encapsulated in canisters designed to fit into the storage vaults and emplacements previously occupied by the spent fuel canisters. The discussion of major design parameters for the repository in Section 6.4.3 shows that such a plan is well within the capabilities of current technology.

SECTION 2

INTRODUCTION

The purpose of this report is to analyze the technical status of both high-level radioactive waste and spent fuel management systems for commercial nuclear reactors. This analysis is designed to assist the members and staff of the California Energy Resources Conservation and Development Commission in carrying out the tasks mandated by AB #2822, as passed by the California State Legislature and signed into law by the Governor in 1976. The complete assembly bill is presented in Appendix D. By this act Section 25524.2 is added to the Public Resources Code (relating to energy conservation); the portions of Section 25524.2 especially relevant to this report are as follows:

"No nuclear fission thermal power plant, including any to which the provisions of this chapter* do not otherwise apply, but excepting those exempted herein, shall be permitted land use in the state, or where applicable, be certified by the commission** until both conditions (a) and (b)*** have been met."

"(a) The Commission finds that there has been developed and that the United States through its authorized agency has approved and there exists a demonstrated technology or means for the disposal of high-level nuclear waste."

Later on in this Section 25524.2 of the California Public Resources Code, paragraph (c) states that "'technology or means for the disposal of high-level nuclear waste' means a method for the permanent and terminal disposition of high-level nuclear waste. It shall not necessarily require that facilities for the application of such technology and/or means be available at the time the commission makes its findings. Such disposition shall not necessarily preclude the possibility of an approved process for retrieval of such waste."

Prior to the Presidential decision of April 7, 1977 to defer reprocessing indefinitely, the National Waste Management plan dealt almost exclusively with the high-level nuclear wastes discharged from fuel reprocessing plants (including cladding hulls and transuranic (TRU) wastes). The major components of this plan which we refer to as the "old mainline HLW program" are: (1) liquid waste storage and containment; (2) solidification of waste by means of spray calcination and in-can melting; (3) immobilization of calcined waste in borosilicate glass (vitrification); (4) encapsulation

* Chapter 196

** ERCDC

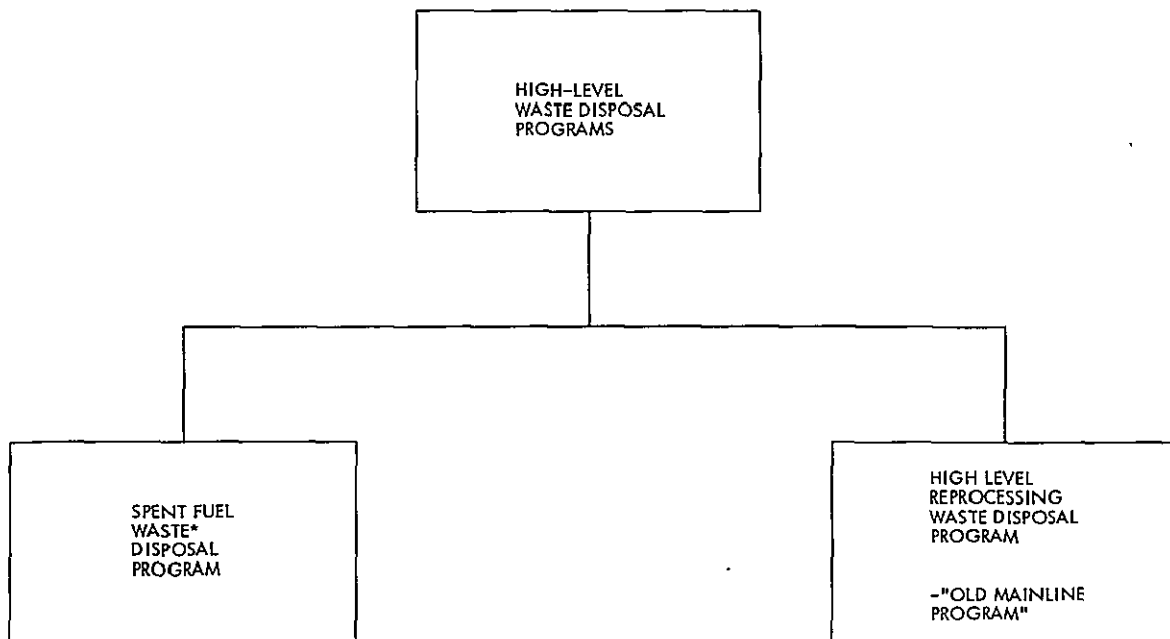
*** Condition (b) requires that the commission report its findings and the reasons therefore pursuant to condition (a) to the State Legislature. It also defines the roles of the Legislature and the Commission after these findings have been submitted.

of waste-glass composite; (5) deep geologic emplacement of canisters in bedded salt for permanent containment and isolation from the biosphere.

In response to President Carter's decision, ERDA is in the process of developing a new program for the storage, packaging and terminal disposal of spent fuel. These two programs are indicated in Figure 2-1. Early on it was decided to limit the present analysis to the "old mainline HLW program", plus the newly-developing program for disposal of spent fuel. Other options are under development (for example, metal matrix rather than borosilicate glass encapsulation of solidified high-level waste; geological emplacement in granite rather than in deep salt beds), but they are not nearly as far along as the components of the "mainline" program. Besides, the National Energy Plan¹ (especially pp.72-73), and recent statements by ERDA representatives² make it quite clear that heavy emphasis will be placed on the newly developing program for spent fuel in an attempt to meet the target date of 1985 for a Federal repository in bedded salt, operating (at first) in a retrievable mode.

A brief and simple account of the origin, nature and disposal options for high level waste contained in Appendix A is designed to give the non-specialist reader an introduction to the present state of high level nuclear waste management.

In order to respond to the directives of paragraphs (a) and (c) of Section 25524.2 (AB 2822) one must first define what is meant by "demonstrated technology or means for the disposal of high-level nuclear waste."



* IF SPENT FUEL IS DECLARED TO BE WASTE.

Figure 2-1. Major U.S. High Level Waste Disposal Program

Many definitions of these terms are possible. Section 3 of this report discusses the terminology and methodology used in the present analysis. Each major component of both waste management systems is analyzed in terms of its Scientific Feasibility, Technical Achievability and Engineering Achievability. The resulting matrix analysis leads to a systematic identification of major unresolved technical or scientific questions and/or "gaps" in the programs, taking into account the important qualifications contained in paragraph (c) of Section 25524.2*.

Before applying the methodology developed in Section 3 to the systems for HLW and spent fuel management, it is useful to examine the dimensions of the nuclear waste problem. Section 4 specifies the origin and nature of both HLW from a reprocessing plant and spent fuel, and contains projections of the annual and cumulative discharge from light-water nuclear reactors during the period 1976-2000. Two projections are given: (1) the 1976 ERDA projection; (2) a new "low" ORNL projection based on the slower-than-planned build-up of LWR's and the Presidential decision to postpone spent fuel reprocessing in the U.S. indefinitely. This projection shows a nuclear generating capacity of 300 GW(e) by the year 2000..

Section 5 examines the technological status of the major components of the "old mainline" HLW program. Section 6 performs a similar explanation of the newly developing program for storage and disposal of unprocessed spent fuel. For unprocessed spent fuel management the major components analyzed are (1) interim storage in existing spent fuel pools at reactor sites, including re-racking to significantly increase storage capacity; (2) interim storage in independent on-site or off-site spent fuel pools; (3) interim (10-20 years) dry storage of spent fuel in an engineered surface or shallow sub-surface retrievable facility (SURFF); (4) packaging of spent fuel bundles or assemblies in suitable containers and emplacement of these canisters in a Federal repository in bedded salt for retrievable disposition for a period of time, followed by terminal disposition. Sections 5 and 6 apply the methodology of Section 3 to each major subcomponent and component of the mainline programs for HLW and spent fuel, respectively. In each case the analysis includes a description of the process, its technical and institutional development history, its program status and plans, and a critical discussion of the major unresolved questions concerning Scientific Feasibility, Technical Achievability, and Engineering Achievability.

In Section 7 the generic or system problems are investigated, including transportation and logistics, interim storage capacity, ownership

*The question of "approval" by "the U.S. through its authorized agency" is difficult and complex. The authors are well aware of the strong interaction between criteria and regulations imposed by environmental, economic and social pressures, and the technology needed to meet these requirements. Since this "feedback loop" is examined in Reference 2-3, the present report is purposely limited to the analysis of technologies reportedly capable of meeting almost any "reasonable" criteria and regulations.

of HLW or spent fuel, allocation of charges for interim storage, influence of licensing procedures, and time schedules for the mainline program.

Section 8 contains the conclusions reached by the authors regarding the status of technology "for the permanent and terminal disposition of high-level nuclear waste" (and/or spent fuel), in the context of paragraphs (a) and (c) of Section 25524.2 (AB 2822). Unresolved technical component and overall systems questions are identified, several suggestions are offered as to the methods that could be used to resolve these questions, and some observations are made about the time schedule of the present programs.

REFERENCES FOR SECTION 2

- 2-1 The National Energy Plan, Executive Office of the President,
Energy Policy and Planning, Washington, D.C., April 29, 1977
- 2-2 Statement of Colin A. Heath, Assistant to the Deputy Director,
Civilian Waste Management, Reprocessing and Recycle; Division
of Waste Management, Production and U.S. ERDA. This statement
was presented at the Wisconsin Assembly Environmental Protection
Committee's Hearings on the Transportation and Disposal
of Radioactive Materials, Madison, Wisconsin, May 16, 1977
- 2-3 OSTP Report , An Analysis of the Back End of the Nuclear
Fuel Cycle with Emphasis on High-Level Waste Management,
California Institute of Technology, Jet Propulsion Laboratory,
T. D. English, Project Manager, August 12, 1977

SECTION 3

METHODOLOGY FOR ANALYZING TECHNOLOGY STATUS

The purpose of this Section is to define the terminology, information sources, and the methodology used to analyze the present status of technology for commercial high level waste management. The radioactive waste management system with which we are concerned comprises the whole set of considerations, plans and implementation methods necessary to accomplish a perceived need of society, i.e., the safe, permanent disposal of high-level radioactive waste and/or spent fuel. The "system" is the aggregate of the scientific, technical and engineering capability, working with established governmental regulatory agencies, to assure continuity of management operations with "acceptable" present and future impacts on society. Some insight into the factors affecting the complete system can be obtained by examining the major system components individually. Of course, the understanding of these components must be integrated in order to understand their interaction with the overall system.

3.1 TERMINOLOGY

A very important subset of the overall HLW management system is the technology necessary to construct the necessary facilities for the safe manipulation and ultimate disposal of these extremely toxic materials. In the present analysis, we are concerned with developing an objective approach to characterize the developmental status of HLW disposal technology. In order to do this, it is first necessary to point out the variety of meanings given to the word "technology", and then to adopt an operational definition for this report. Berenano (Reference 3-1) outlines a spectrum of meanings of "technology".

- In popular conversation "technology frequently pertains to machines and their use."
- "The application of knowledge to practical purposes."
(Reference 3-2)
- "The organization of knowledge for practical purposes."
(Reference 3-3)
- "a system of interrelated innovation, some technical and some social, which comprise some coherent nexus pertaining to the systematic manipulation of the environment." (Reference 3-4)
- "the totality of the means employed by a people to provide itself with the objects of material culture."
(Reference 3-2)

- "systems of rationalized control over large groups of men, events, and machines by small groups of technically skilled men operating through organizational hierarchy" (Reference 3-5).

Hence, there is a very wide range of possible definitions for "technology." This range complicates the interpretation of the term "technology" as used in AB 2822. Some clarifying language is provided by the California Legislature (Reference 3-6).

"As used in this section, 'technology or means for the disposal of high-level nuclear waste' means a method for the permanent and terminal disposition of high-level nuclear waste. It shall not necessarily require that facilities for the application of such technology and/or means be available at the time the commission makes its findings. Such disposition shall not necessarily preclude the possibility of an approved process for retrieval of such waste."

In general, technology incorporates accepted or demonstrated engineering knowledge and experience. This engineering knowledge, in turn, is based on a scientific background which is broadly accepted by the scientific community. In this report the term "technology" is used to mean tools, machines, and basic and applied knowledge necessary for safe, permanent disposal of high level waste. However, since technology is intended for use, its status can be impacted by a change in the manner that society judges it, for example by changing a regulation. In this report the detailed regulatory requirements are treated as a set of external constraints on the high level nuclear waste management technology. The major reason for not formally including a complete set of regulatory requirements in the evaluation of the status of the high level waste management technology is that much of the applicable EPA & NRC criteria and standards are still being developed. If all of these standards and criteria were available during our analysis, they would have been explicitly included in the evaluation of the overall system.

A complete technological system frequently incorporates the use of many component technologies. For example, an airplane is composed of airframe, propulsion system, avionics, etc. Each of these components may be the result of specific technologies. The technological components of a system are the individual sub-features or steps of a technology which are usually chosen to be, as nearly as possible, self-contained steps or unit operations. In the airplane example, once the overall system requirements have been formulated, and the individual performance and interface requirements for the system components, i.e. airframe, propulsion system etc., have been specified, the individual components can be developed in relative independence of one another. Hence it is desirable that such components have a minimum interaction with the satisfactory operation of other components. If a major change occurs in the attainable performance of one of the system components, e.g., the propulsion system, then the performance and interface requirements of the other system components must be reexamined. In many cases modifications must be made to the system component requirements in order to optimally adjust the overall system to the change in the performance of the propulsion system. It should be noted that if the impact of the

change in the attainable performance of a system component is too large, it may not be possible to continue the project viably within the given cost, schedule & performance constraints. However, in many cases it is possible to switch from one form of component technology to another. In the case of HLW management, for example, if the encapsulation of calcined HLW in borosilicate glass should prove to be unsatisfactory, a new technology component such as a "metal matrix" would have to be developed, with in principle minimum impact on the packaging, transportation and disposal of the waste.

3.2 STAGES OF TECHNOLOGY DEVELOPMENT

The sequence of the development of a technology often begins with an intuitive design concept following an analogy in what is thought to be an allied field. Although it is useful to consider that technology progresses sequentially through a number of stages of maturation, it should be kept in mind that in actual programs, the stages may overlap. The amount of overlap between the various developmental stages often depends on the amount of programmatic risk that the program management chooses to take. Frequently, the evaluation of programmatic risk is done on a "seat of the pants basis" rather than through use of a more quantitative "decision analysis" methodology. The basis for judgment of the "success" of the program management is quite often its "batting average." The level of acceptable batting average depends on the norms for a given managerial environment. In the case of permanent disposal of high level nuclear waste it is certainly desirable to reduce the health and safety component of risk to an extremely low level. Hence, the risks associated with large amounts of overlap among the technological stages of the components of the HLW disposal program should be ones involving only timing and economics, and not health and safety.

In the method of technology status evaluation here presented, the following principal stages are used to express the status of the technology for each of the major components of a technology.

- Scientific Feasibility Stage
- Technical Achievability Stage
- Engineering Achievability Stage

Some features which identify these three main stages of technology development are given in Table 3-1.

In the first stage of orderly reconstruction of a technology, it is recognized that there must be a discovery or a recognition of a potentially useful phenomenon or interacting set of phenomena. This is often based on the recognition of opportunity or need, usually supported by theory. In the next stage of development of Scientific Feasibility the discovery is shown to be compatible with recognized physical laws, and an adequate data base either exists or can be constructed. Such a data base, made by experimentation and objectively evaluated, going into depth and detail beyond that which is needed to verify the theory, is absolutely essential in order to be able to develop a sound design concept. The development of a scientifically

feasible design concept, albeit not bounded by economics or elegance, is shown in Table 3-1 as the fourth feature of the Scientific Feasibility stage of technology development.

Following the assurance that a design concept is scientifically feasible, the next step is to determine if it is technically achievable. Referring again to Table 3-1, we have used the following five features to characterize this stage: embodiment of design concept; search for compatible materials for operation; reference design on laboratory scale for proof of concept; development of prototype; design and execution of simulation tests.

The proposed design concept needs to be particularized so that the basic interaction of its phenomenological components can be demonstrated. This idea is expressed as "Embodiment of Design Concept" wherein there can be a demonstration of a specific design suggested in the Scientific Feasibility stage. The specific design at this point is still a demonstration of a phenomenon rather than an application. The next steps in Technical Achievability are to insure the existence of compatible materials for operation, and the preparation of a reference design on a laboratory scale as a proof of concept. Both of these steps are not usually bounded stringently by economics or considerations such as durability. When these steps have been taken, it is appropriate to develop a prototype, or a number of isolated subprototypes of individual aspects of the design concept. Next, using the prototype, it is necessary to devise and carry out simulation tests. These are often based on such theory as existed in Scientific Feasibility work. These simulation tests sometimes lead to extension of theory, whereupon the whole procedure to this point may be iterated for improvement. Indeed, at any point in either the Technical Achievability stage or the Engineering Achievability stage there may be occasion to cause a feedback and iteration into the Scientific Feasibility stage for further work.

When simulation and tests of prototypes or sub-prototypes have been carried out, it is appropriate to investigate Engineering Achievability. However, it should be recalled that in many programs, some aspects of Engineering Achievability may be carried out in parallel with the work on Technical Achievability. In a particular developmental program the degree of overlap between these stages is determined by features such as the "program sense of urgency," resources, and acceptable risks. The entries shown in Table 3-1 under "Engineering Achievability" begin with "Laboratory Demonstration of Integrated System Operation." Such laboratory demonstrations may be performed on a small or breadboard scale, but should include all elements of the system if this is feasible. A key point in progressing through the steps of Engineering Achievability is maintainability. Maintainability means that each element of the system, and the system itself, operates in accordance with design specifications, with no more than a certain maximum acceptable amount of "down-time", and with attendant costs for repairs and replacements. The requirements for maintainability become more stringent through the remaining steps of the Engineering Achievability stage. These steps comprise Pilot Plant, Industrial Scale or Field Trial, and Operating Plant operations. Economic requirements become increasingly stringent as either the system or the sub-system progresses through these last three stages.

Table 3-1. Features of Stages of Technology Development

1. Scientific Feasibility

Scientific suggestion or discovery, usually supported by theory

Compatibility with recognized physical laws

Existence or construction of adequate data base

Proposal of design concept

2. Technical Achievability

Embodiment of design concept

Search for compatible materials for operation

Reference design on laboratory scale for proof of concept

Development of prototype

Design and execution of simulation tests

3. Engineering Achievability

Laboratory demonstration of integrated system operation

Maintainability

Pilot plant

Industrial scale or field trial

Operating plant

The operating plant demonstration is deemed satisfactory when it fulfills the system goals more economically than could be done by alternative technologies. This economics analysis includes consideration both of capital costs and of operational and maintenance costs. In today's environment increasing emphasis is being given to the concept of energy effectiveness. By "energy effective" one means, for example, that the energy expended in the entire nuclear waste management system should be only a small fraction of the net energy output of the reactor system over its lifetime.

The course of development of the Engineering Achievability of a technology is often affected by an iterative improvement process similar to that described in Technical Achievability. Problems arising in the steps of engineering achievement may lead to a revision of the technologies developmental status to the Technical Achievability stage or in some extreme cases to the scientific feasibility stage. This type of unsuspected major problem has impacted several major developmental programs such as the F-111 & Apollo, even though very extensive analysis had been performed.

3.3 SYSTEM REQUIREMENTS

A method for analyzing Technology Status requires that the purpose of the system in which the technology applies be clearly delineated. Such purposes are stated, first in general terms, such as "contain and isolate nuclear waste from the biosphere," "ensure public health and safety," "preserve the environment." The intent of these statements is that the system should meet societal needs while imparting only an acceptably small adverse impact on other co-existing systems. At the second stage these purposes are translated into technical requirements of the system, and are given in more specific terms. In the case of nuclear waste management, they may be grouped into requirements related specifically to the individual system options under consideration. In this report, two systems are examined--management of high level radioactive waste from reprocessed spent fuel rods, and management of unprocessed spent fuel rods. These two systems have some markedly different features, examples of which are listed below:

- (1) The requirements for handling, transportation and the mechanics of emplacement of high-level waste call for solidification, immobilization, and encapsulation, while intact unprocessed fuel rods require only suitable packaging for either interim or permanent disposal.
- (2) The requirements for developing procedures for handling and surveillance of the waste will probably be specific to the system chosen.
- (3) The requirements for organizational responsibility and accountability are quite different for HLW systems and for spent fuel systems, because of the problem of accounting for plutonium in reprocessed waste.

3.4 INFORMATION SOURCES

A key consideration in assessing technology status was the need to obtain the latest possible information. Technical journals and reports although indispensable, are inadequate in themselves because of unavoidable time-lags. To be aware of the current status and capabilities of technology development in a given area, personal visitations constitute an essential part of information gathering, but only if they are supported by a historical knowledge based on the above-mentioned technical journals and reports. There is also value in following non-technical journals

and publications to be aware of developments external to technology that may affect its future status. A list of conferences, workshops, and hearings that were either visited or attended by members of the nuclear waste management team follows:

- (1) International Symposium on the Management of Waste from the LWR Fuel Cycle Conference at Denver (TAD)
- (2) Nuclear Waste Management Systems Overview Workshop (SIO)
- (3) Public Policy Issues in Nuclear Waste Management, Conference at Chicago (CHIC)
- (4) The Tucson Symposium on Waste Management (U of A)
- (5) Workshop on Issues Pertinent to the Development of Environmental Protection Criteria for Waste Management (EPA W)
- (6) Sandia Laboratories (WIPP)
- (7) Sandia Laboratories (Seabed)
- (8) Pacific Northwest Laboratories (PNL)
- (9) Nuclear Regulatory Commission (NRC)
- (10) Idaho National Engineering Laboratory (INEL)
- (11) American Physical Society (APS)
- (12) Electric Power Research Institute (EPRI)
- (13) Energy Research and Development Administration (ERDA)
- (14) Environmental Protection Agency (EPA)
- (15) National Aeronautics and Space Administration (NASA)
- (16) Office of Management and Budget (OMB)
- (17) Oak Ridge National Laboratories (ORNL)
- (18) California Energy Resources Conservation and Development Commission (CERCDC)
- (19) Office of Science and Technology Policy (OSTP)
- (20) Council on Environmental Quality (CEQ)
- (21) Office of Technology Assessment (OTA)
- (22) General Accounting Office (GAO)

- (23) David Rose (MIT)
- (24) Harvey Brooks (Harvard)
- (25) Theodore Taylor (Princeton)
- (26) Terry Lash, Natural Resources Defense Council (NRDC)
- (27) Hannes Alfven (UCSD)
- (28) Lawrence Livermore Laboratories (LLL)
- (29) General Electric, San Jose (GE)
- (30) Boeing, Seattle
- (31) RAND
- (32) Utilities Waste Management Group

To ensure that the entire nuclear waste management team had a broad exposure to both the technological and regulatory issues, a number of people were sent to the key sources of information described above. A summary of the personnel who visited these sources is shown in the matrix in Figure 3-1. The La Jolla Nuclear Waste Management Workshop, Jan. 77, was organized by the JPL/CIT/SIO nuclear waste management team. It enabled the study team to glean from many experts their views of what the major issues are, and what decisions are needed. In addition, the hearings held by the California Energy Research, Conservation, and Development Commission on high-level waste management and reprocessing served as a valuable source of information for this report. These activities, in addition to the general experience of the study participants, provided the study with a broad view of present policies and the range of possible policies to be implemented. This knowledge, in addition to the understanding of Federal programs supplied by our study for the Office of Science and Technology Policy (Ref. 2) was used in this analysis.

3.5 GENERAL ANALYTIC APPROACH

The features of the general analytical approach used in this study are shown in Figure 3-2. These include:

- systems component descriptions
- individual component analysis, and a
- systems analysis

The first group, "System Component Description," requires that the overall system purpose and goal be set forth in terms of functional requirements, including economic parameters, and that the operation of the system be closed and specific. Then the system is separated into components which are as far as possible independent of other components, and the individual

functional requirements of each component are described. The purpose, after the individual component analysis is completed, is to construct a matrix of stages of technology development vs. system component.

In the second group, "Individual Component Analysis," each component has its history and plans laid out, and an examination of existing and proposed criteria is made for each component. From a review of the history and compatibility with the criteria, one can determine the status of technological development at which unanswered issues exist. The analyst can identify pertinent actions needed to resolve the open issues at this point. Simultaneously, the analysis may show further unanticipated open issues, which can be iterated and again carried through the categorization as shown in Figure 3-2. Following this step, the individual component technology stage characterizations are combined into the matrix, as shown in Figure 3-4.

This matrix is used in the "System Analysis" section of Figure 3-2 where the issues related to the technology of each component are identified in each matrix cell. It is at this point that a conclusion can be reached which can declare the technology to be operable or which is the basis for developing a program for action to advance the technology.

In this report, the following stages are used to assess the status of the technology for each of the components of waste disposal systems: Scientific Feasibility, Technical Achievability, and Engineering Achievability. Each of the stages are subdivided further as indicated in Table 3-1 so that a tabulation of sequential and increasingly developed technology can be made. The technological status for each component of a particular waste disposal system can be specified by identifying the state of technological advancement within this framework.

The final decision of the California Energy Commission concerning the existence of a demonstrated technology for high level waste management depends on a consensus among the Commissioners concerning which of the stages must have been completed before technology may be said to exist for each of the major components. It is evident that sincere and experienced individuals could insist, with suitably chosen historical precedents, that the completion of the work for the first, or second, or third of these stages could constitute a demonstration of existence of technology, depending upon the background, experience and interests of the individual. For example, some engineers believe that the technology for disposal of high-level nuclear waste in bedded salt deposits now exists, because more than 25 years of analysis, and a certain amount of field testing experience have not uncovered any problem that cannot be solved with existing technology, including the associated testing and monitoring techniques. Certain equally experienced groups from other fields of work conclude that safety factors and risk assessments will not be satisfactorily understood until many years of operating experience with actual waste packages have been achieved. In assessing a systems developmental status it is worthwhile to delineate a system's operational goals, and then to determine how far along in the first, second, or third stages the development must have progressed, for the

PERSONNEL	LOCATION																															
	TAD	SIO	CHIC	UOFA	EPAW	WIPP	SEABED	PNL	NRC	INEL	APPS	EPRI	ERDA	EPA	NASA	OMB	ORNL	ERCDC	OSTP	CEQ	OTA	GAO	MIT	HARVARD	PRINCETON	NRDC	LLL	GE	BOEING	RAND	UTILITIES	UCSD
DR. T. ENGLISH	X	X	X			X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X				X	X			X	X	X
DR. R. CAMPBELL	X	X			X	X	X	X	X				X	X	X	X	X	X		X	X		X	X	X							X
PROF. L. LEES	X	X	X						X				X	X	X	X		X	X		X										X	X
SIR E. BULLARD		X				X	X	X	X					X	X		X	X	X													X
MR. E. DIVITA P.E.		X	X					X	X				X	X		X	X	X		X	X									X		
DR. C. MILLER		X							X		X	X	X	X		X	X	X		X	X					X	X					
DR. T. KUEHN			X		X									X	X	X		X	X		X		X	X								
MR. A. CHOCKIE		X			X			X		X				X				X					X	X	X			X	X			
MR. R. WILSON		X			X				X	X	X			X	X			X	X			X										
MR. N. WILLIAMS		X		X											X																	
MR. E. EDELSON		X						X																				X	X	X		

Figure 3-1. Visits of JPL/CIT/SIO Nuclear Waste Management Team

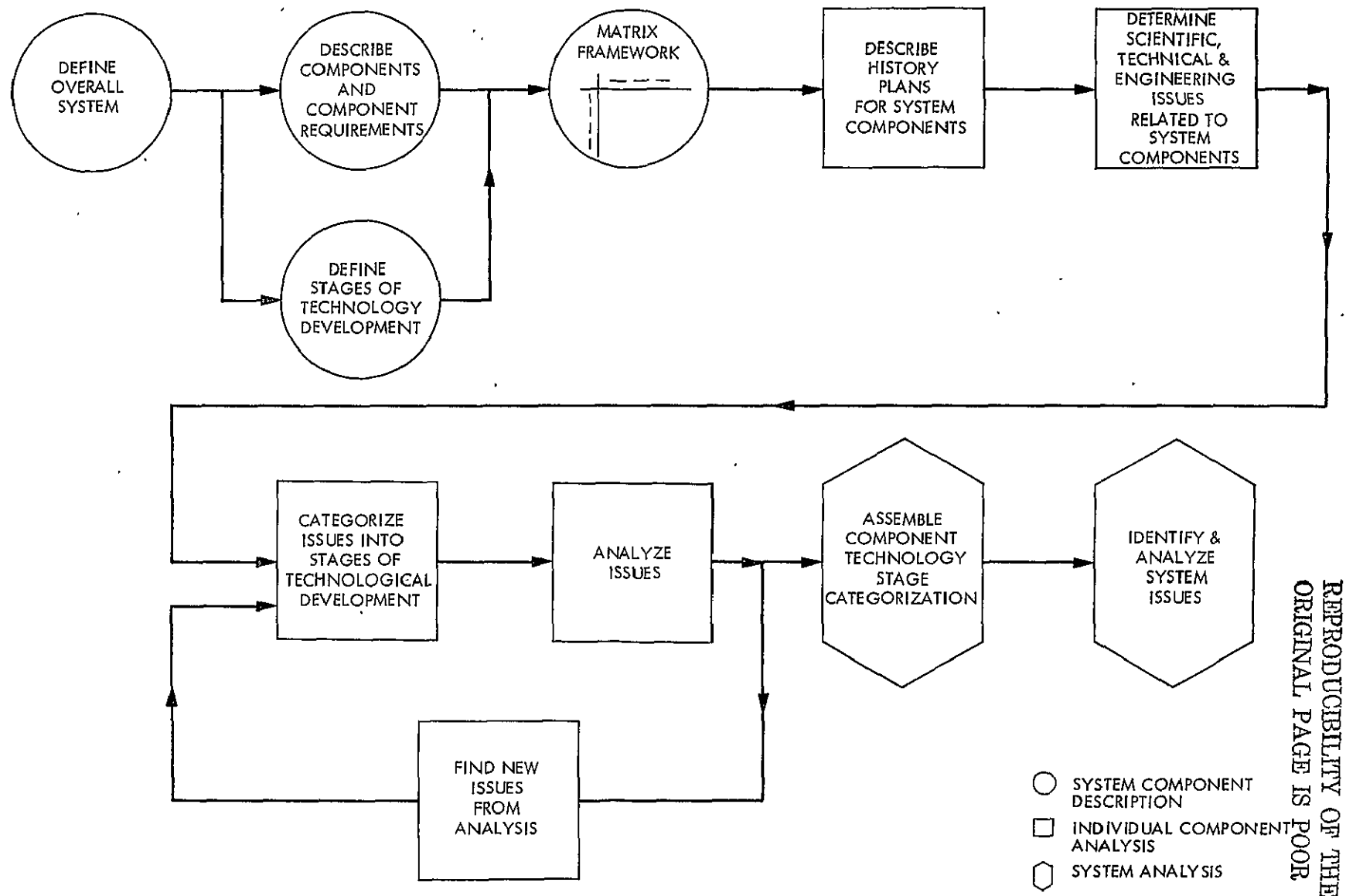


Figure 3-2. General Analytic Approach

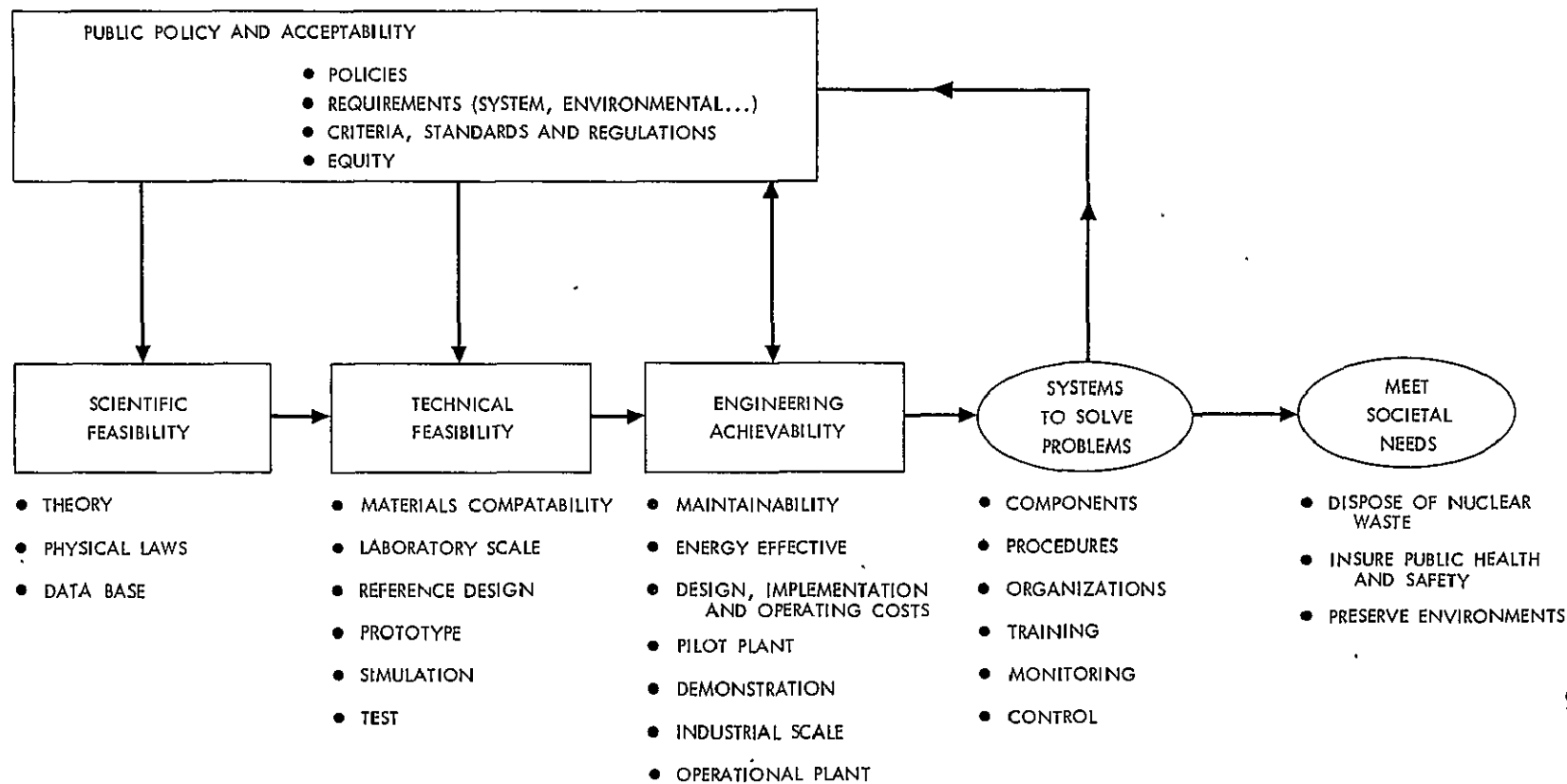


Figure 3-3. Relationship Between Public Policy and States of Technology Development Showing Guidelines

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Stage of Technology Development System Component	Scientific Feasibility	Technical Achievability	Engineering Achievability
Liquid Containment			
Treatment and Packaging			
1. Solidification (calcination)			
2. Immobilization (vitrification)			
3. Encapsulation			
Bedded Salt and Emplacement			

Figure 3-4. Blank Matrix Summary of the Technical Status of Components
in the U.S. Mainline High-Level Waste Disposal System

technology to enable the system's operational goals to be achieved. The result of such an analysis will be a statement, specifying for each component of a system, the developmental status at the time of evaluation. This type of analysis also shows which components of the system need further work, or need to have substitute components provided before the system can be made operative.

3.5.1 Public Policy Constraints

A very important subset of the complete HLW management system is the technological subsystem. Obviously, particular technical methods may exist which are not acceptable because of their potentially deleterious effects on either workers or the general public. Thus, the Scientifically Feasible, Technically Achievable, and Engineering Achievable methods are bounded by public policies and by demands of equity with other systems in the public use. Generally, criteria are set based on environmental considerations such as the need to maintain health or welfare, or on economic considerations such as resource depletion. Figure 3-3 shows the interrelationship between the stages of Technology Development and the constraints on Technology Application which are generally regarded as "Public Policy and Acceptability."

Such dynamic social constraints set requirements which have impacts on the stage of Technology Development, and must be taken into account again each time an iteration takes place as a result of the development of new information. For example, the Nuclear Fuel Services (NFS) reprocessing plant at West Valley, N.Y. did meet regulatory requirements for a period of several years. However, a changing regulatory environment, including both more stringent seismic requirements and for waste solidification requirements, made the NFS plant uneconomical to operate; and it is now closed down.

3.5.2 Identification and Analysis of Issues

For an individual component of a Technology, there may be limitations on its operation due to consequences which arise from its use. Hence the examination of the individual components may reveal features that have unacceptable impacts. The question of acceptance and justification of each potential impact must be considered for each stage of technology development. Issues may arise from conflicts among the interests and values of individuals, groups or institutions. In addition, issues may be technological gaps or unresolved technological problems. These gaps or problems may be characterized according to the categories of Scientific Feasibility, Technical Achievability, or Engineering Achievability. An arrangement for displaying the status of major unresolved issues is outlined in matrix form in Figure 3-4. In using this matrix, key issues would be displayed in the appropriate matrix elements. In order to understand the details of a given identified issue, the reader would have to examine the appropriate section of this report.

The analysis of the issues consists of considering the possible results of using a proposed method to resolve each identified technological problem. Such results may have an impact on later steps or stages of the technology being developed. For example, a lower fraction of radioactive isotopes encapsulated in the glass during HLW solidification will increase the number of canisters required for waste disposal and decrease the radioactivity and decay heat per canister. These consequences must be carefully analyzed in order to close the "loop."

3.6 REFERENCES FOR SECTION 3

- 3-1 Technology as a Social and Political Phenomenon, Phillip L. Berenano, John Wiley and Sons, Santa Barbara.
- 3-2 Webster's Third New International Dictionary (Unabridged), Springfield, Mass., C. Merriam, 1966, page 2348.
- 3-3 Emmanuel G. Mestheme, "The Role of Technology in Society: Some General Implications of the Program's Research," Fourth Annual Report, 1967-1968, Harvard University Program on Technology and Society, page 44.
- 3-4 United States, National Academy of Sciences, Committee on Science and Public Policy, Technology: Processes of Assessment and Choice, Report to the Committee on Science and Astronautics, U.S. House of Representatives, Washington, Government Printing Office, 1969, page 123.
- 3-5 John McDermott, "Technology: The Opiate of the Intellectuals," New York Review of Books, July 31, 1969, page 29.
- 3-6 State of California, "Public Resources Code 25524.2(c)."
- 3-7 "An Analysis of the Back End of the Nuclear Fuel Cycle With Emphasis on High-Level Waste Management," JPL, Pasadena, CA, August 12, 1977.

SECTION 4

OVERVIEW OF HIGH LEVEL WASTE AND SPENT FUEL SYSTEMS

4.1 ORIGIN AND NATURE OF SPENT FUEL

Nuclear power reactors are fueled by uranium dioxide (UO_2) which has been enriched to approximately 3% U-235 (97% U-238). The nuclear fuel is fabricated into pellets (about 1 cm in diameter and 1 inch long) which are then sealed in stainless steel or zircalloy tubes. In the case of boiling water reactor (BWR), the tubes or rods are 4 m long. For pressurized water reactors (PWR) the fuel rods are 3.85 m long and 1 cm in diameter (Reference 4-1, page 2.32). The rods are firmly bound together in square arrays or bundles. The number of rods per array varies depending on the reactor type and vendor; 7 x 7 or 8 x 8 for BWRs and 15 x 15 or 17 x 17 for PWRs (Reference 4-2, page 257). These bundles, when placed in the core of the reactor and surrounded by a flow channel, are defined as assemblies.

Often in the literature concerning the generation of electricity by nuclear power, a reference reactor is described. A PWR reactor used for reference is assumed to produce 1000 megawatt electrical (MWe). The fuel rods in this reactor are capable of a maximum burn-up of 33,000 MWd/MTU after 1100 days of continuous operation. The reference reactor is considered to have a capacity factor of 80% (Reference 4-3, page 2). It should be noted that a reference reactor is a convention used in generic analysis and does not correspond to actual experience. For instance, the average capacity factor exhibited so far by all reactors lies between 60% and 70% (Reference 4-15, page 54 and Reference 4-16, page 55; see also the postscript to the debate between Hans Bethe and Barry Commoner on Nuclear Energy in the Cornell Review, Spring 1977, page 45).

In the reactor core, fission products are produced when the fissile material splits in the fission chain reaction. Most fission products are radioactive. Those products with short half-lives decay rapidly in the fuel and reach an equilibrium point at which production of new atoms essentially equals the loss by decay and burnout. (An example is I-131 which reaches equilibrium in approximately 40 days.) Longer lived fission products (i.e., strontium-90) continue to absorb neutrons to a point where it interferes with the chain reaction. As a result the fuel assemblies are generally removed from the core after 1100 days even though there remains a significant amount of fissile (fissionable) material in the fuel rods (equivalent to about 50% of the original quantity, 243 kg U-235 and 254 kg Pu) (Reference 4-4, page 9). Each year, about 1/3 of the fuel in the core of a light water reactor (BWR or PWR) is removed, and the remaining partially spent fuel is redistributed in the core. This represents about 30 MTU (70 PWR assemblies, 156 BWR assemblies) discharged annually from a 1000 MWe plant (Reference 4-5).

At the time of discharge the irradiated fuel produces almost 180 megacuries of radiation and 1.5 megawatts of thermal power per metric ton of fuel (Reference 4-4, page 18, 19, 20). As shown in Figures 4-1 and 4-1A the heat and radiation levels of the irradiated fuel decay

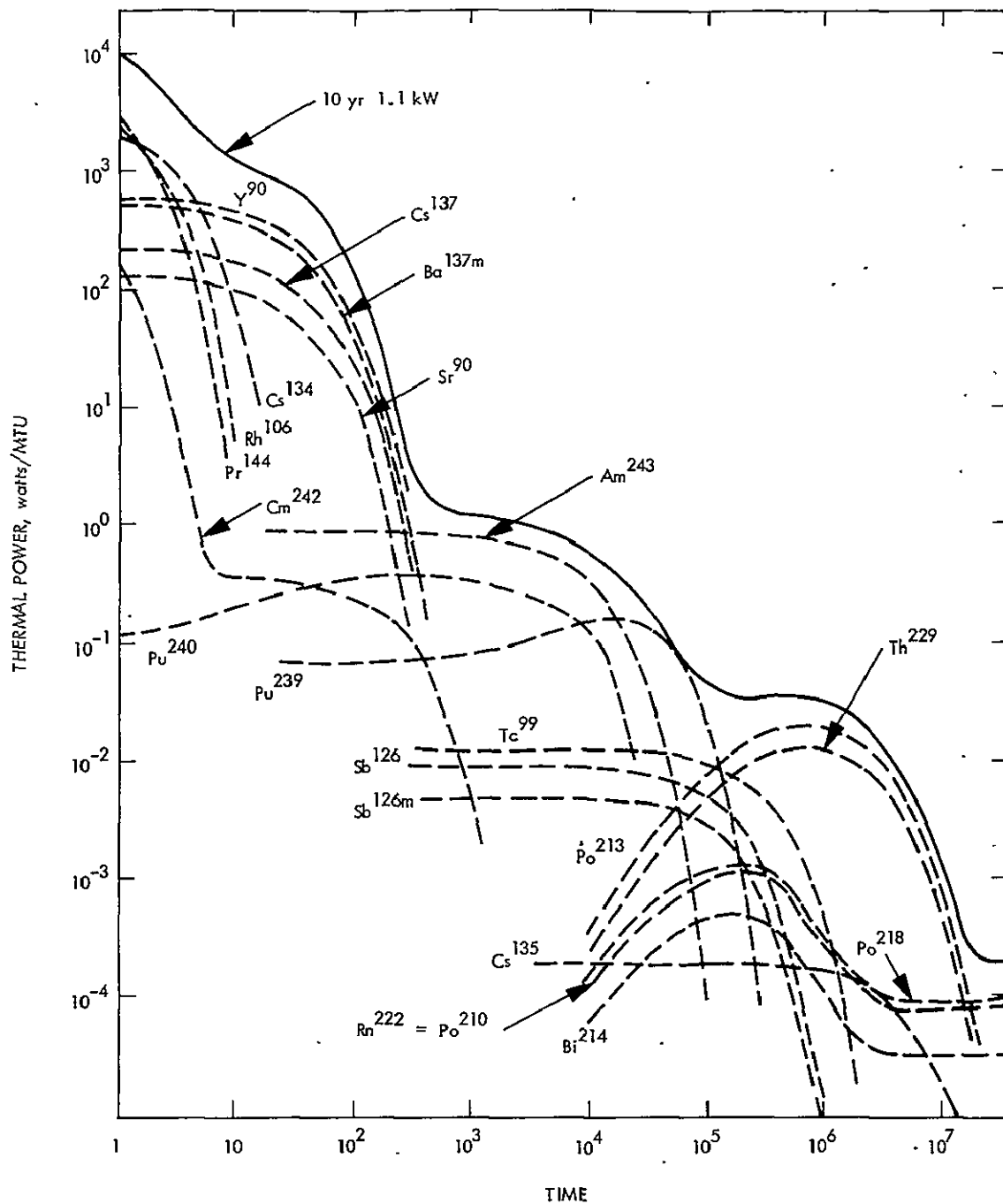


Figure 4-1. Thermal Power Released by the Radioactivity of High-Level Waste from Reprocessing One Metric Ton of Irradiated LWR Fuel Reference 4-3

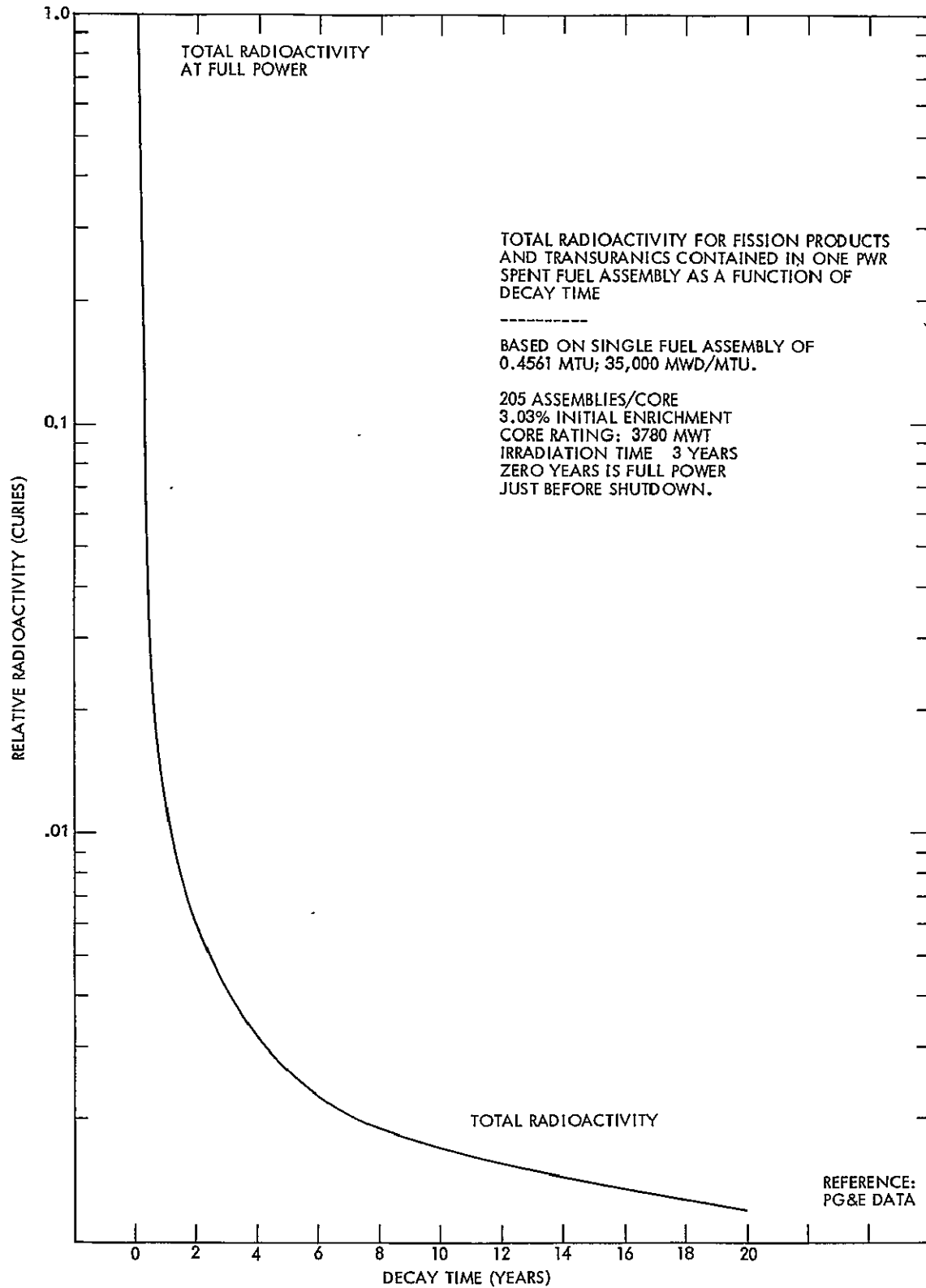


Figure 4-1A. Total Radioactivity for Fission Products and Transuranics contained in one PWR Spent Fuel Assembly as a function of decay time

quickly with time after shutdown of the reactor. For example, 150 days after a discharge the isotopes in the fuel rods containing one metric ton of fuel have decayed to 4 megacuries and have a reduced thermal energy of 20 kilowatts (Reference 4-4, page 18, 19, 20).

4.2 ORIGIN AND NATURE OF HIGH LEVEL LIQUID WASTE

High-level liquid waste is currently defined in 10 CFR 50-Appendix F as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated waste from subsequent extraction cycles, or equivalent, in a facility for reprocessing reactor fuels" (Reference 4-6). The effluent of the solvent re-generation process and the concentrates from the acid recovery system are, in some cases, included in the high-level waste.

The liquid high-level nuclear waste from a reprocessing plant is expected to consist of almost all the fission products (radioactive and nonradioactive) in the spent fuel, a majority of the transuranium actinides formed in the reactor, less than one per cent of the uranium and plutonium, and a large volume of nonradioactive chemicals added during reprocessing. The fission product content of the waste depends essentially on the specific power and burn-up of the fuel assemblies. In the case of the reference PWR with a specific power of 30 MWth/MTU and a burn-up of 33,000 MW-days/MTU, the waste contains approximately 29 kilograms of (non-volatile) fission products per metric ton of fuel (Table 4-1) (Reference 4-7, page 7). This is equivalent to 99.9% of all the nonvolatile fission products in the fuel (Reference 4-7, page 7). Although attempts are made to recover all the uranium and plutonium, there is an expected loss of 0.5% to the high-level waste stream, 0.1% to the aqueous raffinate, 0.01% to the solvent stripping operation (appearing in the solvent regeneration stream), and less than 0.1% going to the cladding waste (Reference 4-4, page 39). Of the remaining transuranic elements, between 90 and 100% are discharged into the high-level waste stream (Reference 4-8, page 30). As revealed in Table 4-1, the quantity of plant corrosion products (nickel, chromium, and iron) and the reprocessing chemicals (hydrogen, nitrate, and phosphate) make up respectively 1.4% and 65% of the weight of the high-level waste.

4.3 RADIATION AND HEAT PROPERTIES

The decay energy of the waste nuclides is such that special considerations must be made for their safe handling, storage, and disposal. Figure 4-1 shows the thermal power of the waste resulting from the reprocessing of one metric ton of fuel. Strontium-90, cesium-137 and their daughters are largely responsible for the high heat levels in the first 400 years after reprocessing, decreasing by three orders of magnitude in this period. (Reference 4-3, page 8). The americium and plutonium isotopes dominate for the next 800,000 years with their decay daughters prevailing thereafter (Reference 4-3, page 8). The substantial amount of heat per metric ton of fuel will dictate the type of measures that can be employed in the "early" years of storage and disposal.

Table 4-1. Materials in High-Level Liquid Waste(a)

GRAMS/MTU from Light-Water Reactor ^(b,c)							
Reprocessing Chemicals		Fuel Product Losses ^(d,e)		Transuranic Elements ^(e)		Other Actinides ^(a)	Total Fission Products ^(f)
Hydrogen	400	Uranium	4,800	Neptunium	480	0.001	28,800
Chromium	200						
Iron	1,100	Plutonium	40	Americum	140		
Nitrate	65,800						
Nickel	100			Curium	40		
Phosphate	900						
Sub-total	68,500		4,840		660		
TOTAL	103,000						

- a. Most constituents are present in soluble, ionic form.
- b. Water content is not shown: all quantities are rounded.
- c. U-235 enriched FWR, using 378 liters of aqueous waste per metric ton, 33000 MWd/MTU exposure.
- d. 0.5% product loss to waste.
- e. At time of reprocessing.
- f. Volatile fission products (tritium, noble gases, iodine and bromine) excluded.

Although the waste heat is of concern, it is the extremely high levels of radiation and the toxic nature of the nuclides in the waste that necessitate the long period of isolation from man and the biosphere. When the waste is discharged, the fission products are the major source of radiation, producing more than 99% of the activity (Reference 4-4, page 53). Almost 90% of the fission product atoms do not have major impact on long term waste disposal considerations because they have half-lives that are either shorter than 4 years and therefore disappear relatively soon, or longer than 4×10^{10} years and therefore have very low levels of radioactivity. (Reference 4-3, page 7). Excluding the very small quantities of isotopes; samarium-151, holmium-166, and selenium-79; there are no fission products with half-lives between 30 and 10^5 years (Reference 4-3). Because the main fission products have half-lives under 30 years, the total fission product activity decays rapidly with time. Strontium-90 and cesium-137 play a significant role in the "early" years, producing a substantial amount of activity and creating about three-quarters of the calculated total hazard. After 700 years, two actinides, plutonium and americium, become the major sources of radioactivity and hazard in the waste. This situation continues for about the next 20,000 years, at which time radium-226 has accumulated to the point of becoming the dominant nuclide in the waste, as far as toxicity is concerned. The radioactive toxicity of ^{226}Ra in water (ingestion) is 170 times larger than the ^{239}Pu toxicity in terms of the number of cubic meters of water per curie required to dilute the substance to the maximum permissible concentration (MPC). In air (inhalation) the situation is reversed, the radioactive toxicity of ^{239}Pu is 50 times larger than the toxicity of ^{226}Ra .

4.4 PROJECTIONS OF SPENT FUEL FROM UNITED STATES COMMERCIAL LIGHT-WATER REACTORS

Before examining the status of the technology for the handling, storage and disposal of the spent fuel discharge at the reactor, it is useful to look at the size of the problem. Some aspects of the back end of the fuel cycle can be described by the following parameters:

With reprocessing:

- Spent fuel discharge - in terms of tonnes per year and cumulative tonnes
- Cumulative shipments and shipments/year to and from FRPs
- Reprocessed and solidified HLW - in terms of tonnes per year and cumulative tonnes.

Without reprocessing:

- Spent fuel discharged - in terms of tonnes per year and cumulative tonnes
- Annual shipments of spent fuel

4.4.1 Nuclear Energy Capacity and Spent Fuel Discharged

All of the parameters are dependent upon the projected growth of nuclear energy. Figure 4-2 and Table 4-2 show the recent changes in projection of nuclear capacity for the year 2000. The 1977 ERDA projection exhibits an annual growth rate of 9.5 percent. For this analysis, the growth of nuclear capacity is kept the same regardless of the decision about reprocessing.

Figure 4-3 shows the cumulative discharge of spent fuel assemblies through the year 2000. Two projections are shown. The lower curve represents the results of ERDA's planning projection based on ERDA long range goals. The upper curve is based on a recent survey completed for ERDA of both existing plants and those under construction. Over the near term, i.e., to 1986, the latter is considered to be the more accurate projection because it is based on actual industry data.

Figure 4-4 uses the number of PWR and BWR bundles discharged to plot the cumulative metric tons of uranium discharged. This information is extended to the year 2000 in Table 4-3. Using the industry projection, there will be 26,000 MTU in 1986. The ERDA projection for the year 2000 shows that 86,000 MTU will have been discharged.

4.4.2 Transportation

Trucking presently accounts for almost all shipments of spent fuel. The shipping cask carried by each truck can carry 1 PWR spent fuel assembly or 2 BWR spent fuel assemblies. The shipping cask carried by rail can carry 7 PWR assemblies or 18 BWR assemblies. (Ref. 4-3A, page 22.15)

Figure 4-5 and Table 4-4 show the annual number of shipments. The simplifying assumptions used here are that reracked and expanded reactor storage capacity is sufficient until 1985. In 1985 facilities are available for shipment of the annually discharged fuel. The upper curve represents the number of shipments if only trucks are used. The lower curve represents the number of shipments if only railroads are used. The middle curve is the projection by ORNL based on a time dependent a mixture of truck and rail shipments from the operating reactors. Each time the spent fuel is shipped to an additional facility a comparable number of shipments must be made.

Assuming a round trip distance of 3000 miles the total truck mileage, if all casks are shipped by truck, is 56×10^6 miles for the year 2000. This is 0.35% of the 1971 total truck mileage (Reference 4-13, page 5-2). Similarly, if all shipments are by rail during the year 2000, the 7.3×10^6 miles is only 0.01% of the 1967 rail mileage (Reference 4-13, page 5-2). The impact of these relatively small number of shipments on the transportation system could be significant if "special trains" are required for rail shipment and if local regulations preclude trucks from carrying these casks through specified areas.

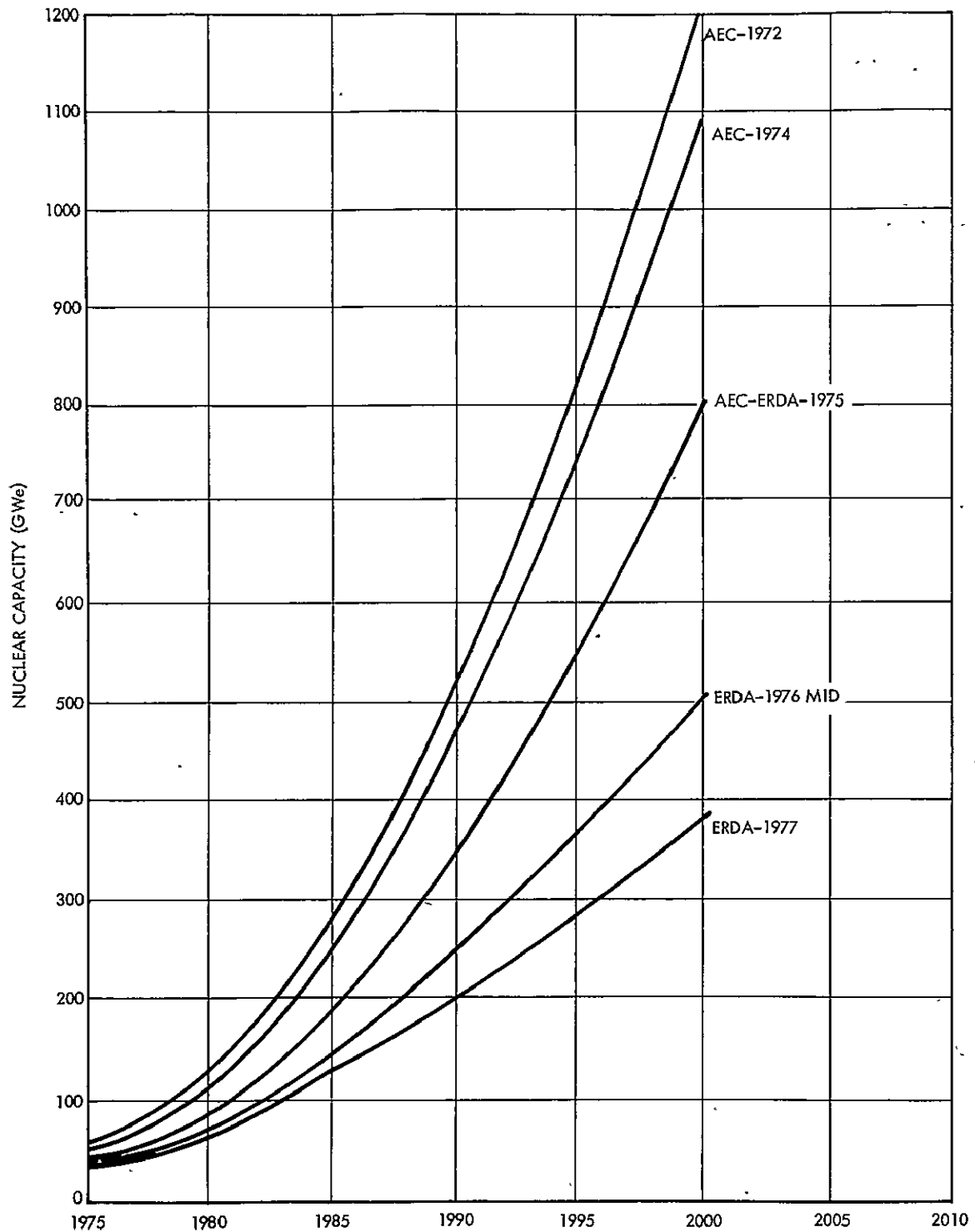


Figure 4-2. Comparison of Nuclear Power Growth Estimates
Source: Reference 4-12, Table 1

Table 4-2. Comparison of Past Forecasts Energy-Electrical Capacity-Nuclear Capacity Year 2000

Total Forecast Made	Total Annual U.S. Energy Consumption (Quads)	Total U.S. Electrical Capacity (GWe)	Nuclear Capacity (GWe)
1972	150	2000	1200
1974	174	2030	1090
1975	174	1750	800
1976	144	1400	510
1977	120	1200	380

4.5 GENERATION OF HIGH-LEVEL WASTE

If the reprocessing option is undertaken, an estimate of the resulting cumulative quantities of solidified high-level waste that will be produced is shown in Figure 4-6. Table 4-6 shows the annual and cumulative quantity of generated waste. This curve assumes that in year marked 0, the first reprocessing plant begins operation at a low capacity factor, that new plants of comparable size (1500 MTU/year) are added every four years, and that each new plant begins operation at a low capacity factor.

4.6 SYSTEMS FOR STORAGE AND DISPOSAL OF SPENT FUEL AND SOLIDIFIED HIGH LEVEL WASTE

The "old mainline program" for high-level waste disposal as examined in this report consists of the most highly developed and potentially available components for the liquid storage, treatment and packaging, and geological disposal of the high-level waste resulting from the operation of a reprocessing plant for spent fuel from commercial nuclear electrical power generation plants. Although other methods of handling the high-level waste are being developed, they are not nearly as close to Engineering Achievability as those components of the mainline, or reference, system for high-level waste disposal examined in this paper. The components discussed in the following sections are:

- (1) Liquid Containment--storage of concentrated commercial high-level liquid waste in stainless steel tanks.
- (2) Treatment and Packaging--solidification of the liquid waste by spray calcination, immobilization of the solidified waste in borosilicate glass utilizing an

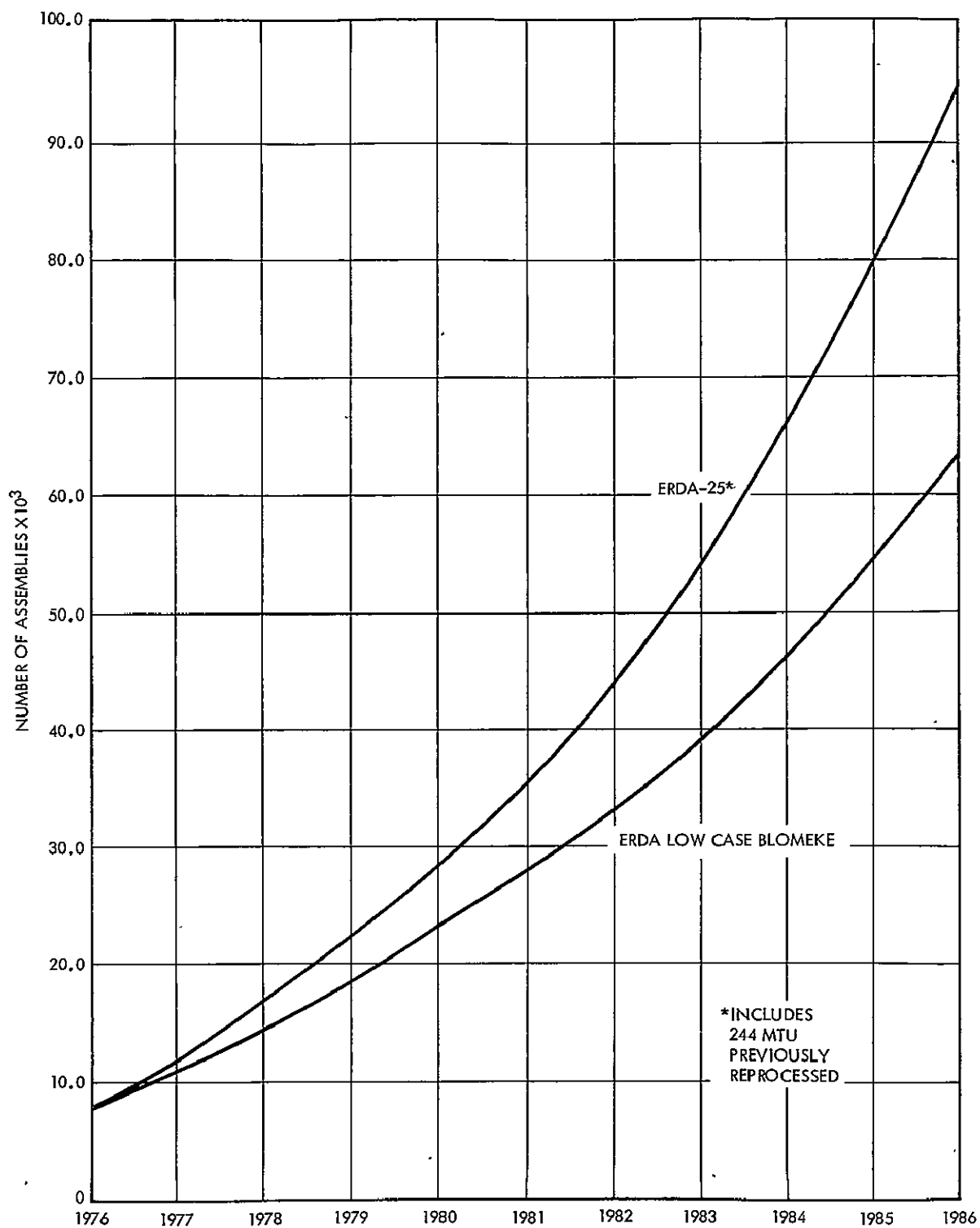


Figure 4-3. Accumulation of LWR Fuel Assemblies
Source: References 4-9 page 15 and 4-11

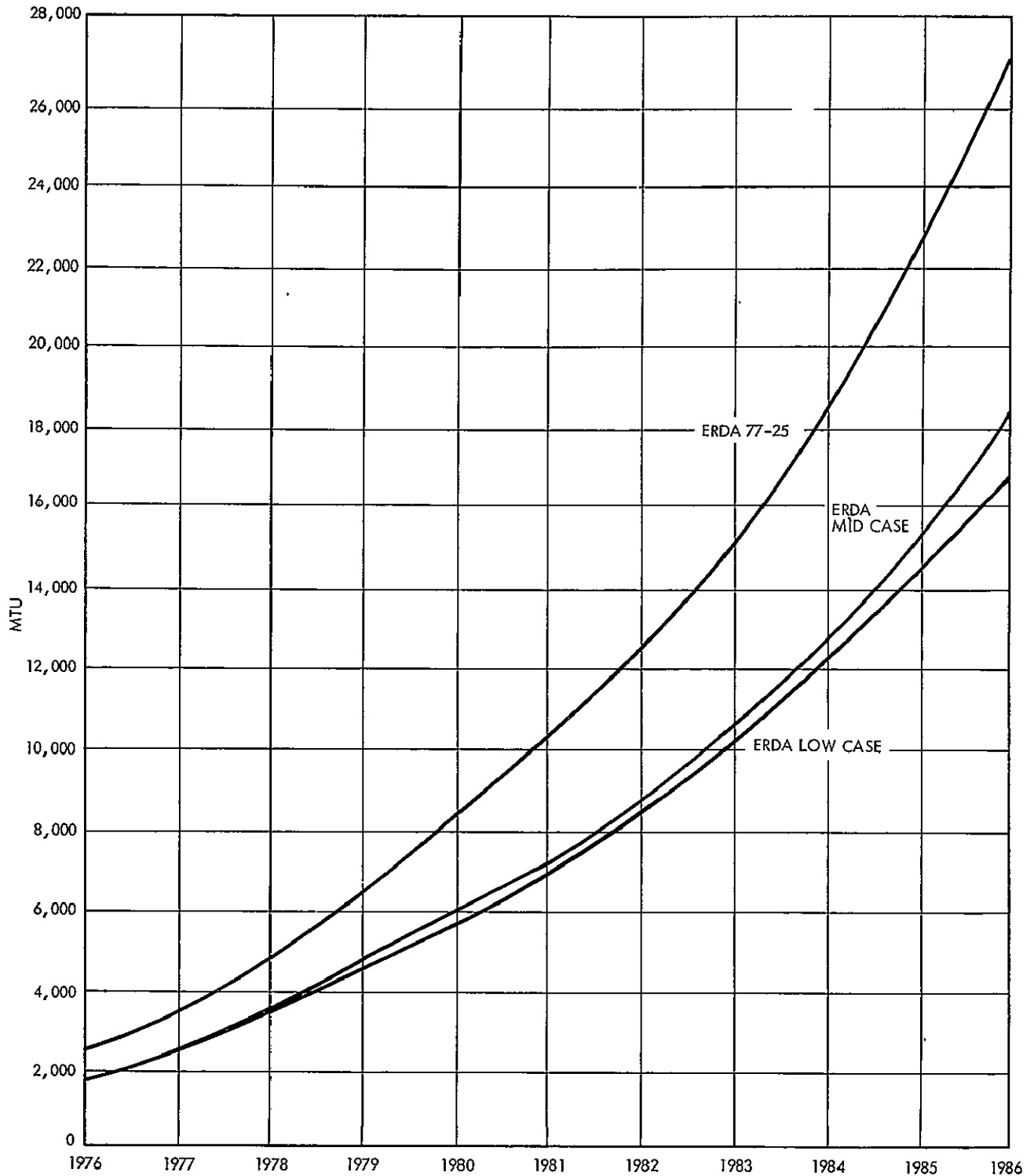


Figure 4-4. Cumulative MTU Discharged
Reference 4-9, page 15, and 4-11

Table 4-3. Cumulative MTU Discharged

Year	Low Case	ERDA 77-25
1970	54	
1971	142	
1972	287	
1973	489	
1974	769	
1975	1197	
1976	1814	2586
1977	2635	3620
1978	3614	4977
1979	4690	6502
1980	5864	8365
1981	7126	10261
1982	8541	12734
1983	10190	15610
1984	12080	18850
1985	14257	22831
1986	16791	27147
1987	19673	
1988	22847	
1989	26292	
1990	30015	
1991	34033	
1992	38365	
1993	43024	
1994	48020	
1995	53743	
1996	59111	
1997	65042	
1998	71762	
1999	78662	
2000	85937	

Assumptions: In Column 2 a ratio of 0.176 MTU/Assembly for a BWR and 0.412 MTU/Assembly for a PWR were used. These were the calculated averages from ERDA-25. Both ratios are slightly lower than those given by Blomeke (Reference 4-3).

In Column 3 the 244 MTU previously reprocessed is included.

Source: References 4-9 page 15, and 4-11

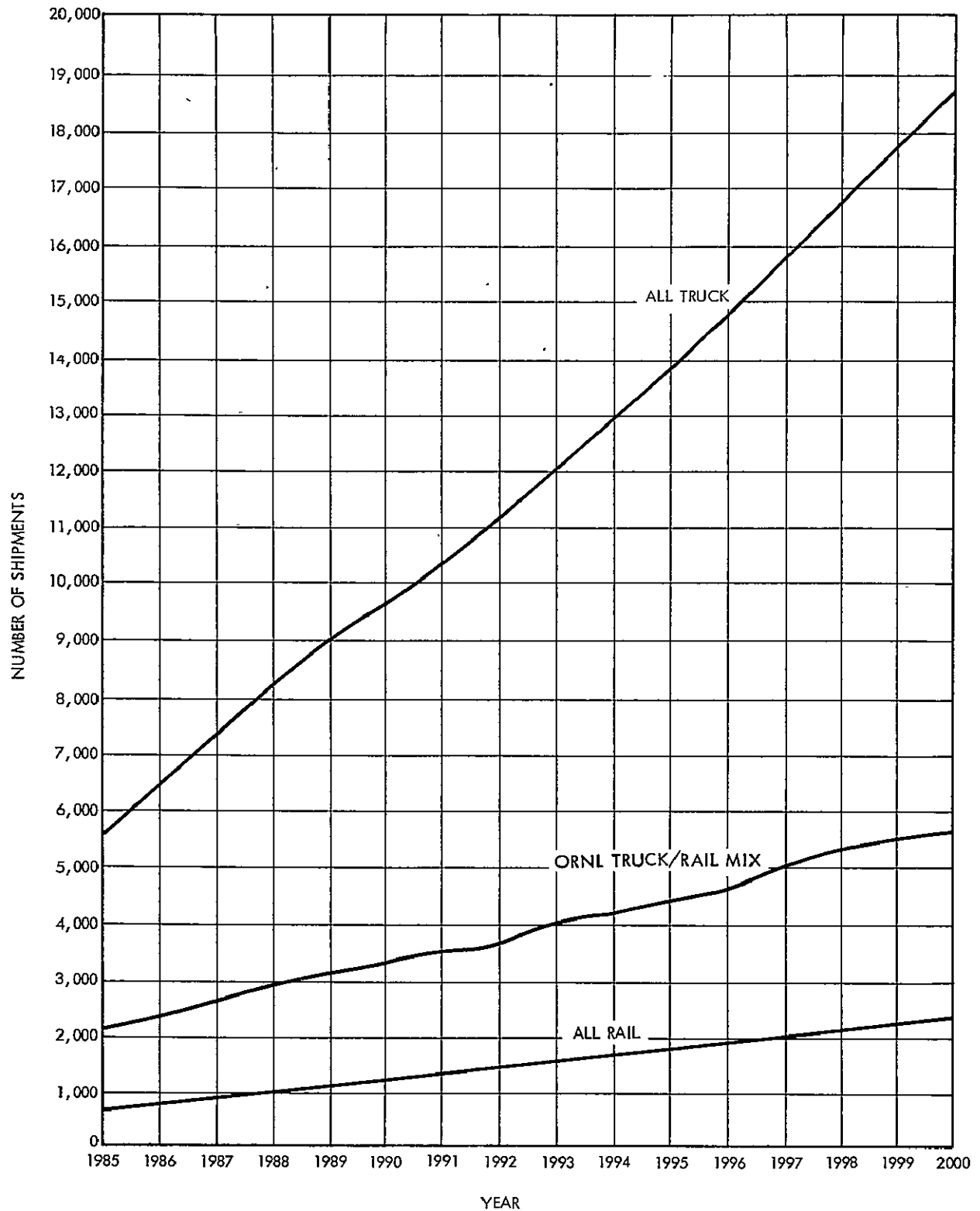


Figure 4-5. Annual Shipment of Spent Fuel
Source: Reference 4-11

Table 4-4. Annual Shipments of Spent Fuel

Year	Shipments						ORNL MIX
	All Truck			All Rail			
	BWR	PWR	Total	BWR	PWR	Total	Total
1985	2,345	3,281	5,626	260	469	730	2,107
1986	2,695	3,853	6,548	299	550	849	2,403
1987	3,113	4,337	7,450	346	620	966	2,670
1988	3,426	4,780	8,206	380	683	1,063	2,940
1989	3,709	5,109	8,904	412	742	1,154	3,115
1990	4,014	5,610	9,624	446	801	1,247	3,289
1991	4,334	6,053	10,387	482	865	1,347	3,550
1992	4,677	6,521	11,198	520	932	1,452	3,733
1993	5,038	7,008	12,046	560	1,001	1,561	4,015
1994	5,411	7,506	12,917	601	1,072	1,673	4,199
1995	5,807	8,033	13,840	645	1,148	1,793	4,499
1996	6,240	8,602	14,842	693	1,229	1,991	4,770
1997	6,679	9,181	15,860	742	1,312	2,054	5,022
1998	7,107	9,756	16,863	790	1,394	2,184	5,211
1999	7,530	10,321	17,852	837	1,474	2,311	5,509
2000	7,948	10,872	18,820	883	1,553	2,436	5,652
Total Shipments			190,982				62,614

Source: Reference 4-11

Table 4-5. Generation of Solidified High-Level Waste (MT)

Year	Number of FRP*	Annual	Cumulative
0	1	500	500
1	1	1500	2000
2	1	1500	3500
3	1	1500	5000
4	2	2000	7000
5	2	3000	10000
6	2	3000	13000
7	2	3000	16000
8	3	3500	19500
9	3	4500	24000
10	3	4500	28500
11	3	4500	33000
12	4	5000	38000
13	4	6000	44000
14	4	6000	50000
15	4	6000	56000
16	5	6500	62500
17	5	7000	69500
18	5	7000	76500
19	5	7000	83500
20	6	7500	91000

*Fuel Reprocessing Plants

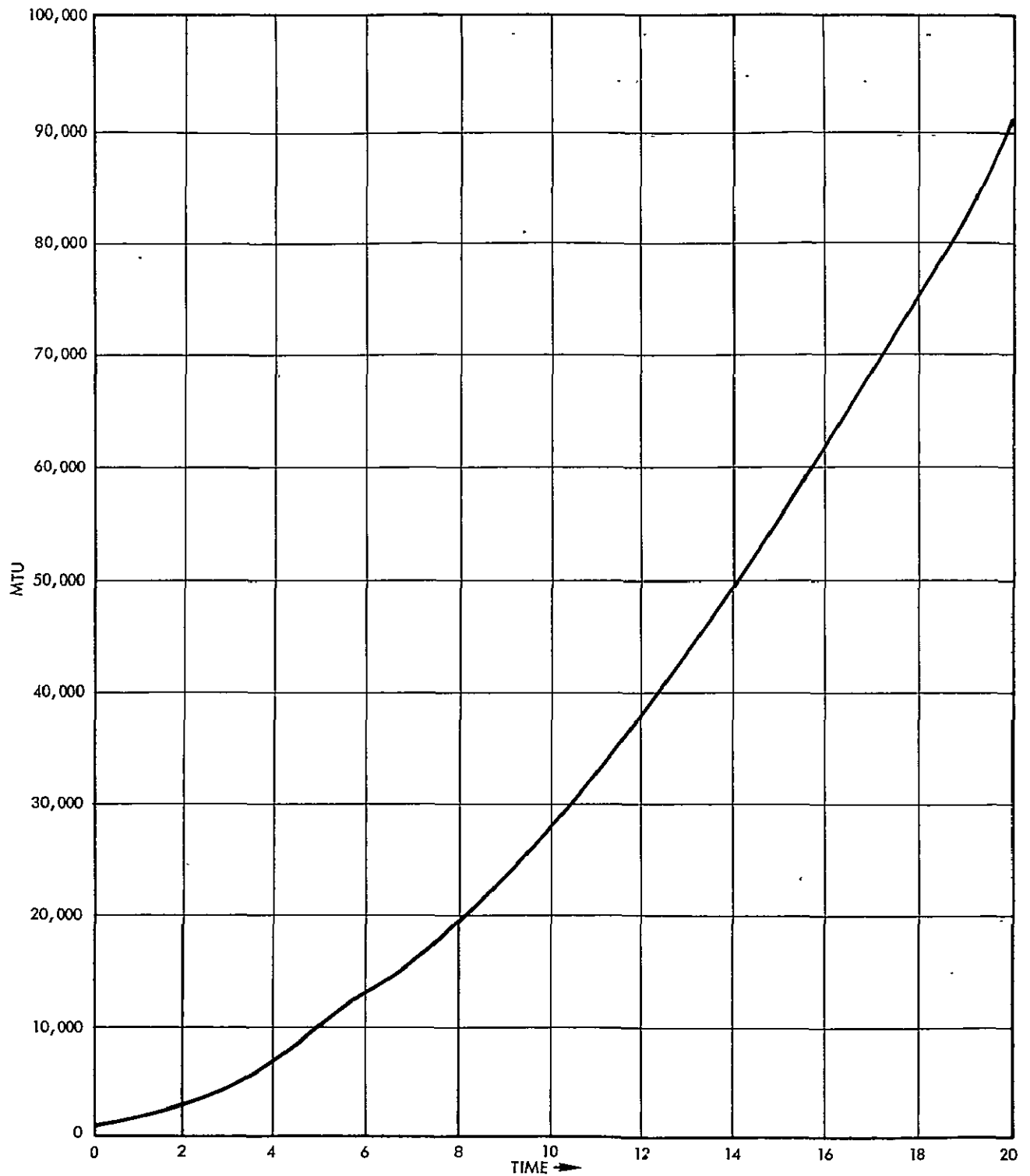


Figure 4-6. Cumulative Liquid High-Level Waste from Commercial Nuclear Power Reactors Versus Time after Opening First Commercial Reprocessing Plant

in-can melter, and encapsulation of the vitrified waste in the storage/disposal canister.

- (3) Deep Geological Disposal--emplacement of the encapsulated vitrified waste in a deep bedded salt formation.

In April 1977, President Carter announced an indefinite postponement of reprocessing operations in the United States. The consequence of this action was to shift attention to extended storage of spent fuel from commercial nuclear power plants. Having developed under the assumption that reprocessing services would be available by the early 1980s, nuclear power plants have been designed with only limited storage capacity for spent fuel ($1 \frac{1}{3}$ to $1 \frac{2}{3}$ core capacity) (Reference 4-14, page 5). The first action by the nuclear plant operators is, and has been, the expansion of existing reactor pools. Two central or off-site spent fuel pool facilities are in the licensing stage. A spent unprocessed fuel facility for dry storage (SURFF) to be operated by the Federal Government is in the initial design phase. Also in the design phase is the method by which spent fuel could eventually be permanently disposed of in a deep geological formation. Specifically, these components for spent fuel storage/disposal are defined for the purposes of this paper as the "mainline program" for spent fuel and are examined as follows (Figure 4-7):

- (1) Reactor Pool Storage--the utilization of all available reactor pool space through reracking while retaining the space for the discharge of the full reactor core.
- (2) Independent Pool Storage--independent, either on-site or away-from-reactor (AFR), spent fuel storage facilities with the capacity and ability to store PWR and BWR assemblies (as originally discharged from the reactor) from numerous reactors for an extended period of time.
- (3) Spent Unprocessed Fuel Facility (SURFF)--an independent engineered surface facility for the dry storage of spent fuel assemblies or bundles that have been packaged in metal containers.
- (4) Deep Geological Disposal--the emplacement of packaged spent fuel assemblies in a deep bedded salt formation.

Options other than those considered in these two programs for HLW and spent fuel are being pursued by ERDA, but at a much lower level of emphasis. For commercial HLW, for example, elimination of liquid containment and immediate solidification is being considered. Fluidized-bed calcination is under development as a possible alternative to spray calcination. Encapsulation of solidified waste in phosphate glass or in a metal matrix is being studied as an alternative to borosilicate glass. Work on options other than deep geologic disposal in bedded salt formations, such as disposal in granite, sea-bed disposal, transmutation, and disposal in space, is continuing in the study and design phases. But it is clear from recent statements by ERDA and the National Energy Plan that the major thrust of the NWM program is along the two lines discussed in this report.

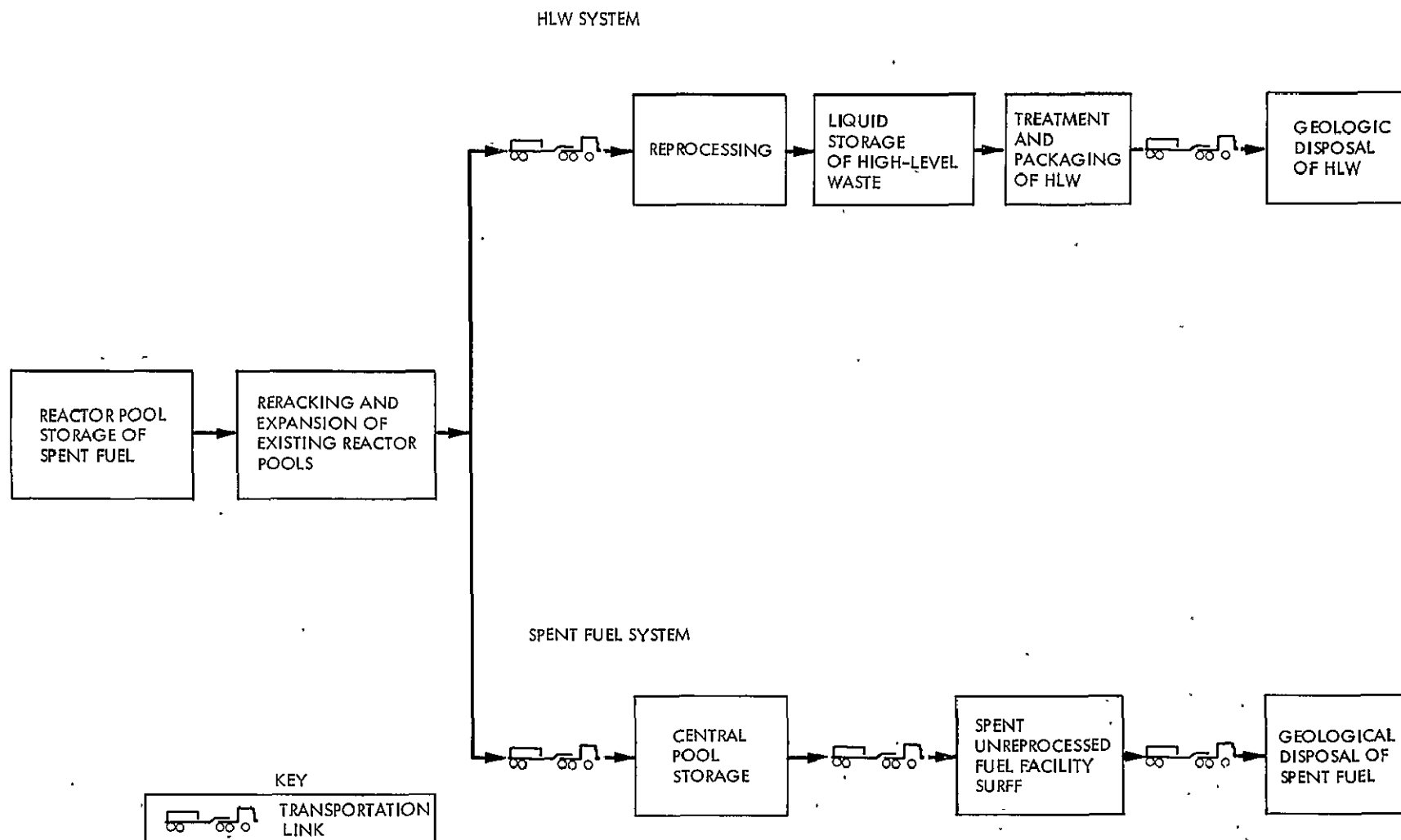


Figure 4-7. Systems of Interest

Clearly not all of the spent fuel will pass through all of the stages shown on the lower path of Figure 4-7. For example, a considerable fraction of the total spent fuel may be shipped directly from spent fuel pools at reactor sites to a packaging facility located at the site of a Federal geological repository. However, because of the present fluid state of the national program all the components described previously (1)-(4) must be analyzed.

REFERENCES FOR SECTION 4

- 4-1 Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle, Vol. 1, ERDA-76-43, U.S. Energy Research and Development Administration, 1976.
- 4-2 American Nuclear Society Transactions. "Spent Fuel Storage, Papers 2 and 5", Vol. 25, June, 1977.
- 4-3 Cohen, Bernard L., "High Level Radioactive Waste From Light Water Reactors," University of Pittsburg, (1976).
- 4-3A Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, ERDA 76-43, May, 1976, page 22.15
- 4-4 Blomeke, J. O., "Projections of Radioactive Waste to be Generated by the United States Nuclear Power Industry," ORNL-TM-3965, (1974).
- 4-5 "Environmental Survey of the Uranium Fuel Cycle," WASH-1248, U.S. Atomic Energy Agency, Washington, D.C., (1974).
- 4-6 Code of Federal Regulations, Appendix F to 10 CFR 50, "Policy Related to the Siting of Commercial Fuel Reprocessing Plans and Related Waste Management Facilities," U.S. Government Printing Office, Washington, D.C., (1972).
- 4-7 "High-Level Radioactive Waste Management Alternative," WASH-1297, Battelle Northwest Laboratories, Richland, WA, (1974).
- 4-8 "Radioactive Waste Management Practices in Western Europe," European Nuclear Energy Agency, OECD/NEA, (1971).
- 4-9 U.S. Energy Research and Development Administration, LWR Spent Fuel Disposition Capabilities, 1977-1986 (ERDA 77-25), May 1977
- 4-10 Kee, C.W., Croft, A.G., Blomeke, S.O., Updated Projections of Radioactive Wastes to be Generated by the U.S. Nuclear Power Industry (ORNL ITM - 5427) December 1976, Oak Ridge National Laboratory, Oak Ridge, Tennessee.
- 4-11 Data supplied by J.O. Blomeke, Oak Ridge National Laboratory, April 1977.
- 4-12 Testimony of James R. Schlesinger, Assistant to the President, Before the Subcommittee on Fossil and Nuclear Energy Research Development and Demonstration of the Committee on Science and Technology United States House of Representatives," June 7, 1977.
- 4-13 General Electric "Fuel Storage Facility Expansion for Morris Operation," (NEDO, 21624) April 1977

- 4-14 See Testimony before the California Energy Resources
Conservation and Development Commission of Keith Schwartztraber,
page 5, March 10, 1977.
- 4-15 Selbin, Joel, "Unreal Thinking About Energy," Bulletin
of Atomic Scientists, September 1977, pages 54, 55.
- 4-16 Bethe, Hans A., "Nuclear Power: A Safe Bet," Bulletin
of Atomic Scientists, September 1977, page 55.

SECTION 5

ASSESSMENT OF STATUS OF "OLD MAINLINE" PROGRAM
FOR DISPOSAL OF SOLIDIFIED HIGH-LEVEL WASTE

In this section we assess the status of current technology development of the various components of the mainline system for the disposal of waste from commercial reprocessing plants.

- (1) Liquid Containment (storage of concentrated high-level liquid waste in stainless steel tanks)
- (2) Treatment and Packaging (solidification of the waste by spray calcination, immobilization of the solidified waste in borosilicate glass utilizing an in-can melter, and encapsulation of the waste)
- (3) Deep Geologic Isolation (bedded salt emplacement)

5.1 LIQUID CONTAINMENT OF HIGH-LEVEL WASTE

5.1.1 Process Description.

High-level liquid radioactive waste streams originate in the reprocessing procedure when the spent fuel rods are chopped into small manageable pieces, and the contained UO_2 fuel along with its combined fission products and fission by-products is dissolved in nitric acid. The solution is chemically treated to separate the bulk of the uranium and plutonium from all other materials. The acid solution, stripped of uranium and plutonium is generally called "High-level Liquid Waste," the nomenclature coming from 10 CFR Part 50, Appendix F.

For some commercial and military wastes (Nuclear Fuel Services, Inc., Idaho Falls and Savannah River) the containment is accomplished by storing the radioactive liquid for an extended period of time in protected tanks to allow the radioactivity and the heat-generating ability to decrease.

Current advanced design, as given in a description of the AGNS process (Reference 5-1) generates about 4000 liters of high-level liquid waste per metric ton of uranium processed. For fuel that is reprocessed at 150 days after removal from a reactor, the waste has about 4 million curies of radioactivity, and about 20 kW of heat per metric ton of fuel processed. The solution is concentrated before reaching the holding tank by a factor of about 6, so that the initial storage receives 600 liters of liquid waste from one metric ton of fuel.

Typically, the non-aqueous content of the high-level waste generated from the treatment of a single metric ton of light water reactor spent fuel is 103 kg. The principal chemical constituents are 34 kg of fission products and actinides, 70 kg of treatment chemicals, 4.8 kg U and 0.05 kg Pu (Reference 5-2). This concentrated liquid initially generates 35 watts per liter, and this heat must be removed at a rate which keeps

the temperature of the container below the design level. For very small tanks, the surface-to-volume ratio is large and convective air cooling will suffice. For larger volume tanks, it becomes necessary to include submerged cooling coils through which cooling water flows. Normal safety features include spare cooling coils on a standby system, spare tanks for complete transfer of content, and operation at atmospheric pressure by means of venting through absolute filters and condensers.

The liquid HLW could be further concentrated, either immediately or during the interim storage period after additional decay of the heat producing nuclides. Economies in the number of tanks needed would be expected. However, chemical limitations on the degree of concentration exist because of the increased temperature of the concentrated solutions and because of the possibility of segregation of solids in the holding tank. Such segregated solids would settle to the bottom of the tank and would constitute a bottom layer of sediment which would not be an efficient heat transfer configuration for transfer of the heat generated in the sediment layer itself to the body of liquid in the tank. If the bottom sediment layer contained a considerable part of the heat-producing radionuclides, there could be local superheating of the liquid at the surface under a pressure head of the liquid height in the tank, resulting in violent bumping due to expansion of superheated vapor.

5.1.2 History of Storage of Liquid Waste

Tank storage of large quantities of chemicals has been developed, both in the United States and abroad, to a state where handling is now accomplished for a wide range of corrosive and difficult chemicals. Examples include corrosives and acids, flammables, and cryogenes.

Storage of high-level radioactive waste solutions in tanks $\geq 1000 \text{ m}^3$ differs from that for the usual corrosive and difficult materials because of the continuously self-generated heat. This self-generated heat may change the corrosion pathways due to the continuously changing chemical nature of the stored liquid. Furthermore, because of the high degree of containment required, it has been necessary to incorporate sensitive methods for remote leak detection and to facilitate self repair methods in the tanks. When the dissolved solids content of the liquid is high, as in concentrated or in neutralized solutions, it is possible by maintaining the air surrounding the containing tank at a very low relative humidity, that liquid seeping through cracks or pinholes in the tank will lose its water through evaporation and the dissolved material in the seepage will precipitate out and plug the crack. This serves as a self-repair method for the tank which contains high solute concentration liquids. Additionally, secondary containment consisting of catch tanks and drip trays, and spare tank capacity to receive the contents of an unsatisfactory holding tank are used as safety measures.

There is extensive U.S. and foreign experience in tank storage of high-level liquid wastes resulting from military reprocessing plants. Some of this experience has been accrued over more than 25 years.

Table 5-1 sets forth the tank storage capacity, as far as is known, for all foreign countries and for the United States. The total U.S. experience is some 100-fold greater insofar as volume is concerned than all the rest of the tankages combined, excluding the [possibly large] amount in Russia, China, and India. In this analysis, the design and operating experience in the U.S. is assumed sufficient to evaluate the technological status of containment of liquid-acid high-level waste.

Table 5-1. High Level Nuclear Waste Storage Capacity in Cubic Meters, 1971.

Location	Capability (Cubic Meters)
W. Germany	
Karlsruhe	100
Julich	6
Great Britain	
Dounreay	400
Windscale	710
France	
Marcoule and La Hague	960
Belgium	
Mol	504
Japan	360 (Planned)
Russia	Not Available
China	Not Available
India	Not Available
United States	
Hanford	247000
Savannah River	67000
INEL (ICPP)	8300
West Valley	5600 (Capacity)
	(327,900 Total)

(Adapted from References 5-3 and 5-4)

Experience in the United States has been to a large extent based on the handling of High-Level Waste from fuel used in military reactors. Military reactor use differs from commercial reactor use in that the fuel is burned to a much smaller degree than commercial reactor fuel, before being reprocessed. This ratio may be 1/3 to 1/10 or even lower, and the content of fission products, and hence heat generation, in military High-Level Waste is correspondingly less.

The earliest production of quantities of high-level waste took place at Oak Ridge National Laboratory in 1944. The burnup, and hence the fission product content was quite low, and the storage was planned for asphalt-lined concrete tanks, two of which were put into use. In the interest of greater safety, a series of larger tanks was built; 6 tanks of 150 m³ capacity, consisting of mild-steel lined concrete tanks, located underground. In order to avoid corrosion of the steel, the nitric acid waste was neutralized with sodium hydroxide, and the resulting material stored in the tanks. It was felt that additional safety accrued to the system because the precipitate formed upon neutralization carried down most of the radioactive fission products as an insoluble sludge. In case of a breach of the tank containment, the insoluble sludge would not sink into the soil but would remain as a solid for transfer to other containment. The actual transfer of radioactive sludge from a breached tank is beset with many difficulties, e.g. difficulty in (remotely) removing caked sludge from around reinforcing ribs in the tanks. However, because the sludge in which the fission products are contained is relatively immobile, in principle the removal can be accomplished without significant time urgency. The supernatant liquid in the tank contained mainly the Cs-137, which does not form a precipitate under these conditions, and would present less of a problem in the soil than would the full range of fission products.

A larger operation was initiated at the ERDA site at Hanford, Washington beginning in 1945 where concrete tanks with mild steel lining were used for containment of neutralized waste, the tanks being of 2000 m³ and 3000 m³ capacity. At a later date, similar tanks were used for low heat generation wastes; less than 0.01 watts/liter. A similar design with the addition of gas-jet agitators in the sludge was used for the larger tanks where the waste ranged up to 8 watts/liter.

During the years 1956-1973 an estimated 420,000 gallons of liquid high level waste containing about half a million curies leaked from the Hanford tanks. These tanks contained a total of 50 million gallons of high level waste. A further modification of design was made so that all tanks built since 1968 have been of double wall construction. Particular attention has been given to the weldment. The metal walls have been heat-treated after fabrication to relieve residual stresses. The three types of tanks are shown in Figures 5-1, 5-2, and 5-3.

The history at Savannah River covers a smaller tankage volume (see Table 5-1). The tanks, however, were generally similar to those at Hanford. The final conclusion of the study of the leaks at Hanford was that the cause of tank leakage in radioactive waste storage service was stress corrosion at welds where metal composition had changed due to the welding. The leaks in the tanks at Savannah River were contained because of the cup-and-saucer configuration of the tank sitting on a secondary containment pan.

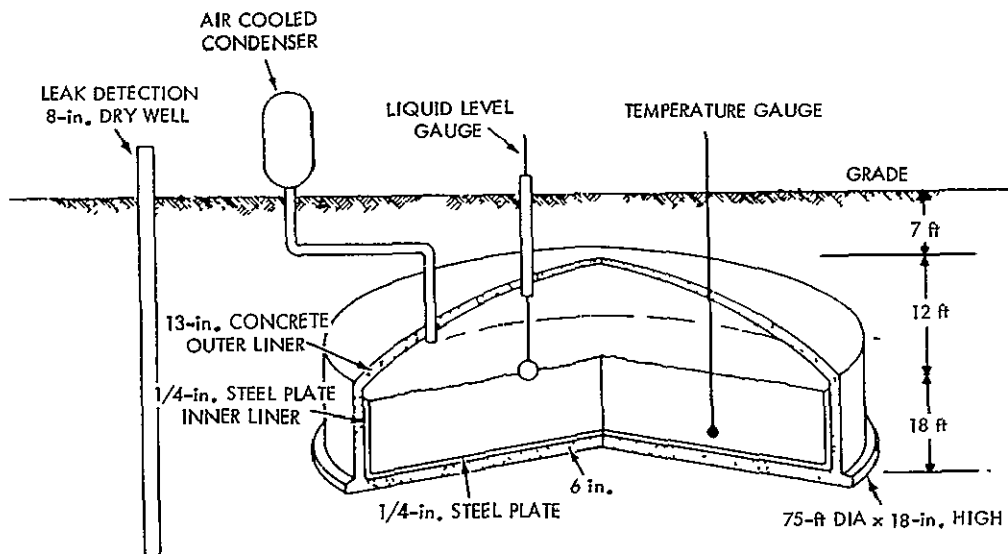


Figure 5-1. Typical Storage Tank for Nonboiling Waste
(From Reference 5-5, pg E-2)

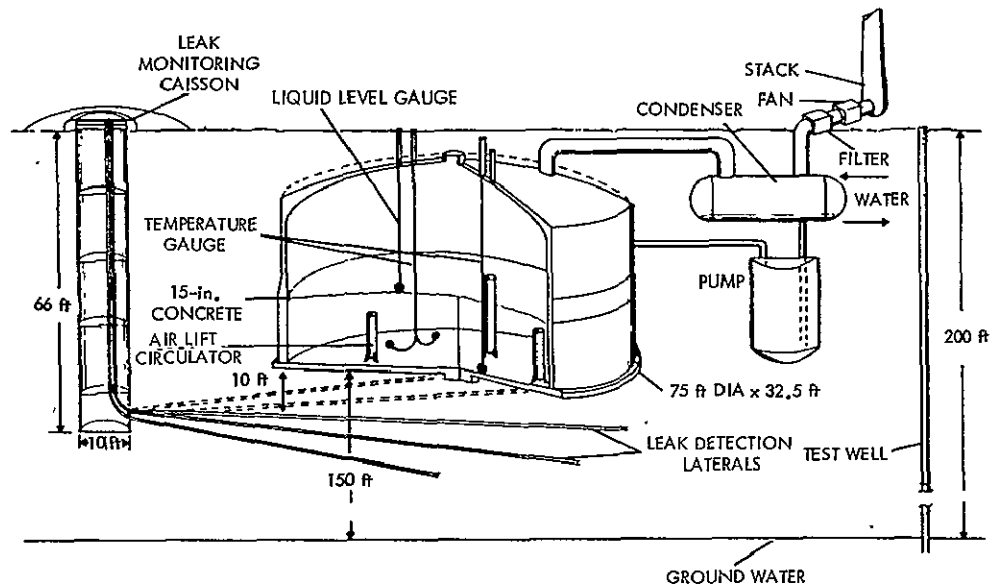


Figure 5-2. Typical Control and Safety Features for High-level Boiling Radioactive Waste Storage
(From Reference 5-5, page E-2)

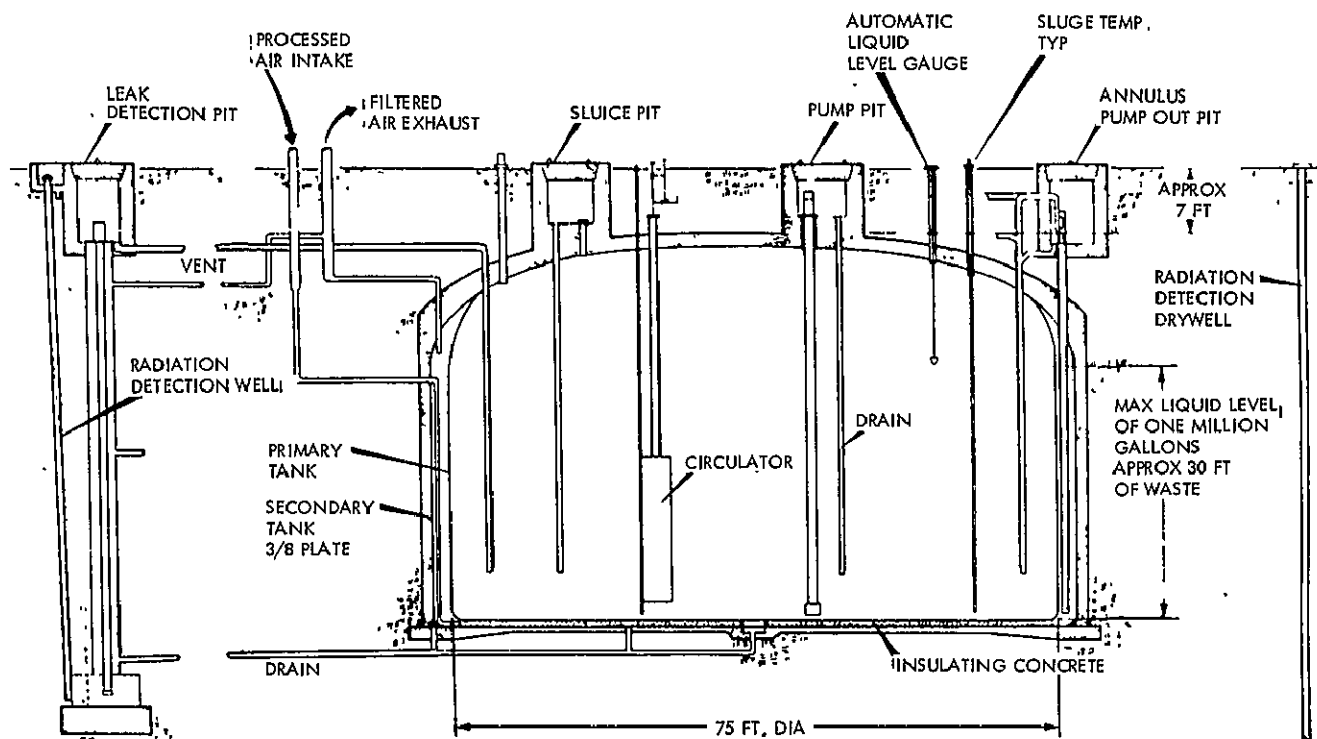


Figure 5-3. Double-Wall Boiling Waste Tank
(From Reference 5-5, page E-3)

The experience of both Hanford and Savannah River was remarkably similar; 12% of the tanks at Hanford leaked, and 14% of the tanks at Savannah River leaked. The Savannah River tanks because of the secondary containment did not result in discharge of waste into the ground. Both of these tank-farms handled low-burnup military waste in the form of neutralized sludge and supernatant liquid, which were thought to be compatible with mild steel tanks.

In 1966, the Nuclear Fuel Services, Inc. plant in West Valley, New York initiated commercial reprocessing activities using high level waste storage tanks of 2850 m³ capacity based on Savannah River design features, namely, in situ stress relief of the welded tanks, the use of a separate catchpan under the complete holding tank, and concrete lined excavation area for tertiary containment. The reprocessing streams have been treated as at Hanford and at Savannah River with alkali in order to neutralize the wastes and permit the use of carbon steel tanks.

To summarize, the operations at Oak Ridge, Savannah River, Hanford, and West Valley with storage of liquid waste used neutralized waste. The containment of neutralized liquid high-level waste is useful from an operational and management perspective, but it does not provide a direct demonstration for the use of acidic high-level waste storage.

Non-neutralized high level storage waste experience has been gained as Idaho National Engineering Laboratories (INEL). The reprocessed fuel at INEL has been left acidic and has been stored in stainless steel tanks. In addition, some of the fuel reprocessed at INEL has a burnup that approaches the burnup of commercial reactors. The experience dates from 1952 when stainless steel tanks were put into service. The acid high-level waste at INEL is held in a number of stainless steel tanks of 1100 m³ capacity together with some smaller tanks of 110 m³ capacity. Altogether, 8300 m³ acid high-level waste are in storage at any one time since an active program is carried out to reduce aged waste to calcine and thus free more volume for fresh waste. A careful management program, covering temperature profile monitoring, corrosion test coupons in-tank to estimate wall corrosion, and the homogeneity of the acidwaste content has resulted in a history of no leaks. (Reference 5-5A, page E-7).

5.1.3 Liquid-acid High-level Waste Containment Program Status and Plans.

The chemistry of HLW solutions during an extended holding period is compatible with storage in selected stainless steels as mentioned above in the discussion of INEL experience. Interest continues on potential effects of excess zirconium content which may result from corroded zircalloy in long-cooling-time fuel bundles. The reason is that zirconia precipitate may act as a scavenger and hinder plutonium separation from reprocessing streams. A continuing program to determine acid concentration limits and temperature limits in preparation for possible future needs is in progress at Oak Ridge National Laboratory (Reference 5-6).

5.1.4 Issue Related to Technology Status of Liquid Containmentment

An early review of the literature related to liquid high-level waste containment identified three issues of concern. These were (1) stability of the liquid during containment, (2) the compatibility of the acid liquid waste and the steel tanks, and (3) the availability of adequate control technology to take care of the off-gases coming from the tank. Based on the current literature concerning acid waste storage, and presentations at CERCDC headings concerning waste disposal means, we consider that the above three issues have been shown, by the INEL experience, to be resolved for military waste. The same methods are applicable to commercial waste if the heat loadings, with which experience has been accumulated, are not exceeded. Furthermore, there appears to be no significant change in the composition of the liquid waste over time that would affect future interim holding operations. The history of storage at INEL and measurements from test-coupons in the tank indicates that liquid acid-waste and the existing stainless steel storage tank are compatible. In addition, the filter and monitoring control systems have been adequate to control the off-gases.

5.2 TREATMENT AND PACKAGING

The mainline system for the solidification of the high-level liquid waste (HLLW) consists of three operations: solidification of the liquid in a spray calciner, immobilization of the calcined waste in borosilicate glass using an in-can melter vitrification process, and finally, the encapsulation of the vitrified waste in the storage/disposal canister.

The solidification of the HLLW by calcination in a spray calciner is intended to satisfy Federal regulations which require the liquid to be converted to a dry solid which is "chemically, radiolytically, and thermally stable" (Reference 5-7). Due to the potential dispersability of the calcined waste, additional containment is obtained through incorporation of the calcined waste product within a suitable material. For the mainline system the waste is incorporated into a borosilicate glass in an in-can melter. The borosilicate waste/glass solid is encapsulated in the formation canister and after going through leak checks and decontamination procedures is eventually planned to be sent to a Federal repository. A schematic diagram of the three phase process is presented in Figure 5-4.

Except in the initial development stage, spray calcination and in-pot melting have been investigated together. Even though calcination, vitrification, and packaging of the high level waste will be operated as an integral process, the issues on the status of operation and development are generally concerned with specific steps of the process.

5.2.1 Solidification of Waste by Spray Calcination

5.2.1.1 Process Description. As shown schematically in Figure 5-5, HLLW is pumped into an internal mixing pneumatic atomizing nozzle located at the top of the spray chamber. The atomized droplets are flash-dried and calcined as they fall through the chamber (700°C wall temperature). The short period of time required for calcination results in essentially no holdup time in the

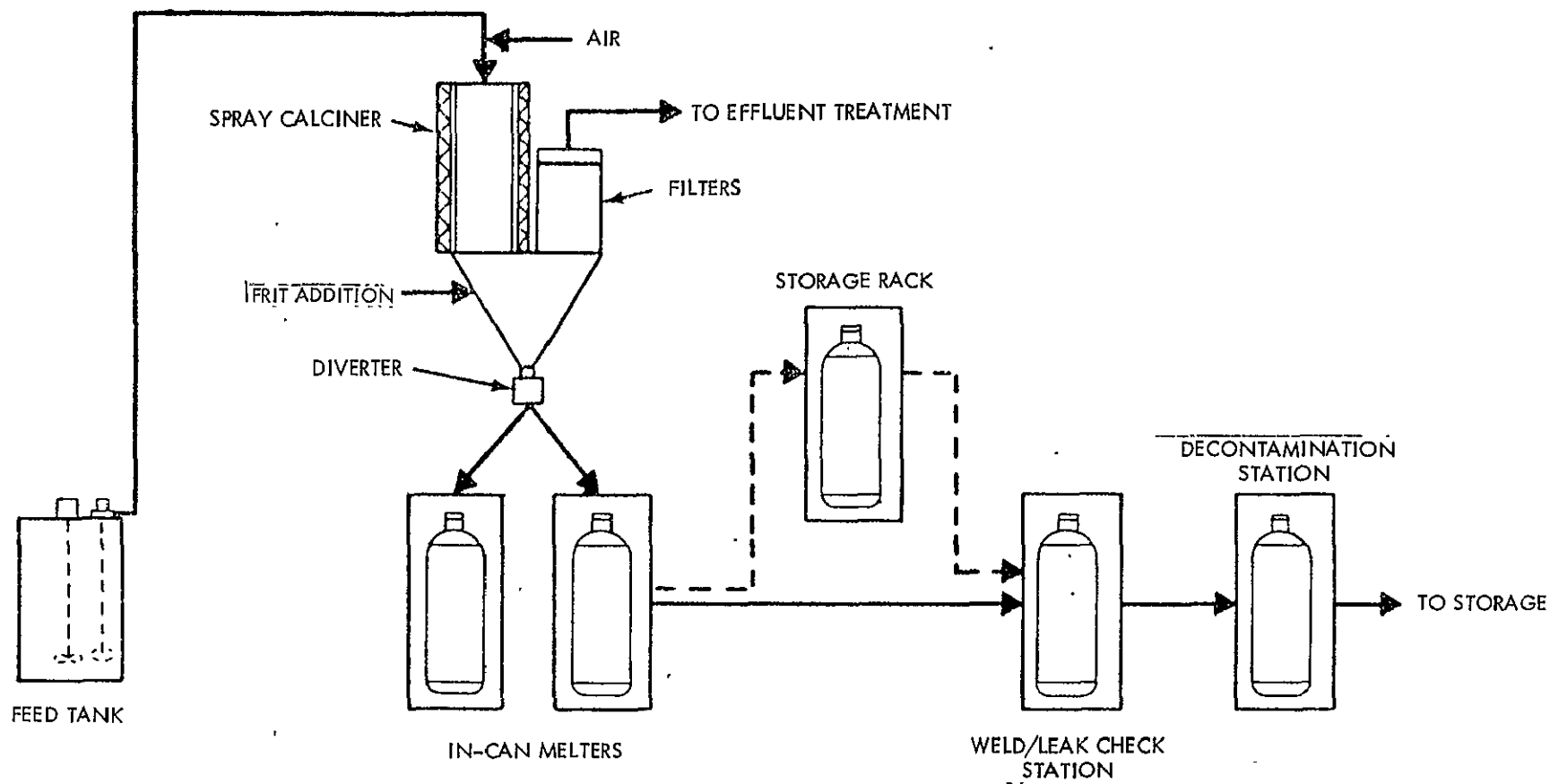


Figure 5-4. High-Level Liquid-Waste Process Flow Diagram (Reference 5-8)

chamber for the calcined product. The powder products (fines) that travel with the offgas are separated from the off-gas by stainless steel filters which are periodically cleaned by reverse direction pulses of air. The pressure drop across the filters determines the time between pulses which may vary from 5 to 30 minutes. The fines are then discharged with the main calcine product. Atomizing air, steam and oxides of nitrogen exit through the filters. With less than 0.1% of the nonvolatile radionuclides passing through the filters, there is no significant load on the effluent treatment subsystem and no major amount of liquid to be recycled to the calciner from the treatment subsystem (Reference 5-8, page 25).

5.2.1.2 Development History. Spray calcination has been in development at the Battelle Northwest Laboratories (BNWL) for over 16 years. The Waste Solidification Engineering Prototypes (WSEP) program, which existed between November 1966 and October 1970 ran both radioactive and non-radioactive tests with various calcination processes, including the spray calcination technique (Reference 5-10). Thirteen fully radioactive engineering-scale operational runs were undertaken on the spray calciner during the WSEP program. Each run lasted for about 50 hours. An additional 1000 hours were obtained using simulated wastes at feed rates approaching those to be used in conjunction with a full scale reprocessing facility - 75 liters/hr versus 80 liters/hr required for a 5 MTU/day (1500 MTU/year) plant (Reference 5-11, page 6.28). Though the anticipated waste product has changed since the WSEP program began, making the WSEP calcined product and phosphate glass forms obsolete, the WSEP program did provide important process operation information on borosilicate glass. In particular it provided verification of both process operability and maintainability (Reference 5-9, page 5).

Since 1972, BNWL has been operating the Waste Fixation Program (WFP) sponsored by the Nuclear Fuel Cycle and Production Division of ERDA. The goal of this program has been to fully develop a process for the conversion of HLLW to a borosilicate waste-glass mixture. The process currently receiving the major development efforts is spray calcination and in-can melting (glass formation) (Reference 5-9, page 6).

5.2.1.3 Program Status and Plans. The Waste Fixation Program (WFP) at BNWL is continuing to develop the spray calciner coupled to the in-can melter. A full scale non-radioactive facility has been operating since April, 1977, and has calcined at a rate up to 200 liters/hr (Reference 5-12, page 1). A full scale remote operation and maintenance non-radioactive facility is now in design that will allow experience to be gained in remote canister handling and maintenance techniques (Reference 5-9, page 20). Radioactive tests using high level waste from commercial irradiated fuel will be carried out in late 1978, if this is decided to be consistent with President Carter's decision to indefinitely defer reprocessing. The pilot plant spray calciner/in-can melter process planned to be used handles 15 liter/hr (Reference 5-12, page 1).

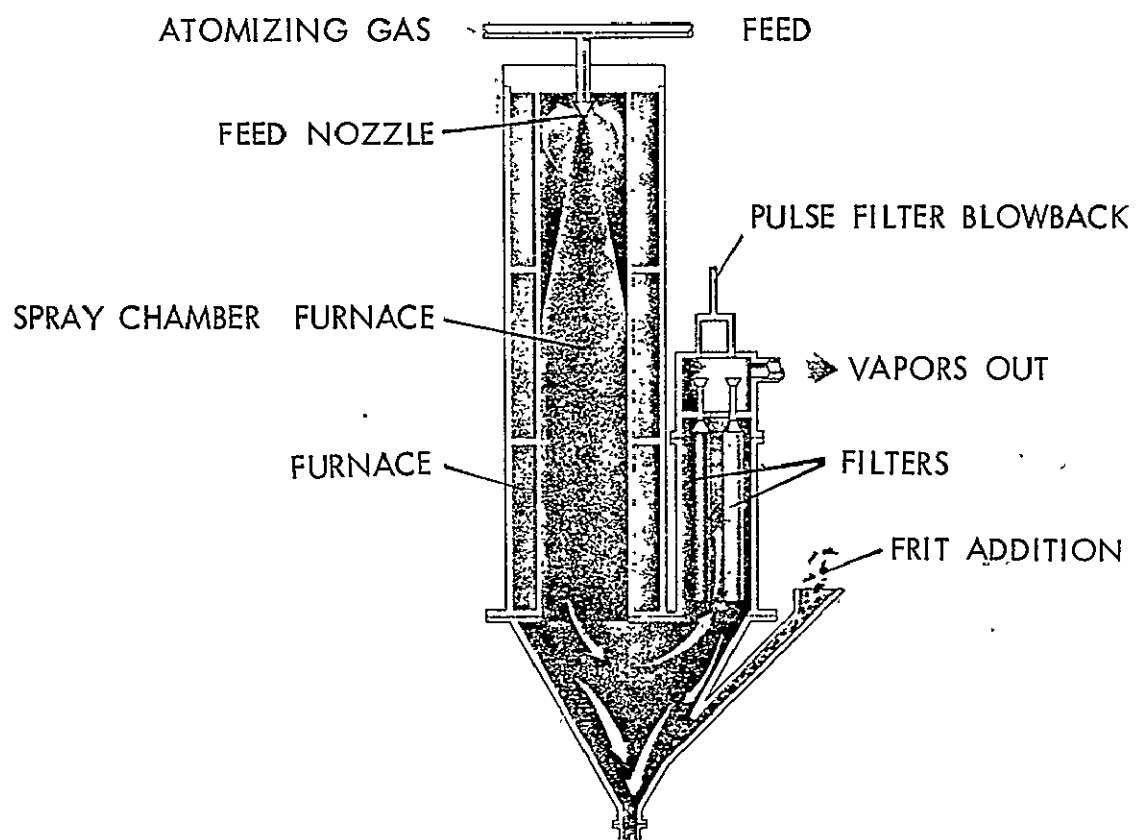


Figure 5-5. Diagram of Spray Calciner (Adapted from Reference 5-8, page 27)

5.2.1.4 Issues Related to Technology Status of Spray Calcination

5.2.1.4.1 Scientific Feasibility

Calcine Integrity and Stability. Federal Regulations require a dry and stable high-level waste form for long-term disposal. The calcine product resulting from the flash evaporation of the waste in the spray calciner chamber consists of particles with a mean diameter of 10 micrometers (Reference 5-8, page 25). This very finely divided product is highly dispersible and if contacted by water, would almost immediately release some of the radio-nuclides to the contacting water - such as strontium and cesium (Reference 5-12, page 1). Although tests have been undertaken to examine the pressurization potential of calcines in sealed canisters subjected to high temperature, the calcine product from the spray calciner is not considered a suitable form for such canister storage or disposal (References 5-11 and 5-13). Other calcination methods, such as fluidized-bed, would be employed which could produce a calcine that is less potentially dispersible and more suitable for containment because of its granular form as opposed to the powder obtained from the spray calciner (Reference 5-14).

The spray calciner has been developed in connection with a melter, and as a result the calcine is immediately converted to glass in this system. Since there is zero holdup time in the spray chamber, the waste is only in a solid dry (fine powder) calcine form for the period necessary for gravity feeding it into the melter and to attain fusion of the calcine-glass frit mixture. Therefore, the issue of a stable solidified form of the waste with respect to the spray calciner should be addressed to the final product of the spray calciner/in-can melter - the vitrified waste (See Section 5.2.2.4).

5.2.1.4.2 Technical Feasibility

Process Operation. Component and total process reliability is an important part of the operation of the spray calciner.

The early development phase of the spray calciner revealed the metallic atomizing nozzle as an item that potentially would require frequent maintenance due to erosion (Reference 5-8, page 28). A ceramic atomizing nozzle was developed about 5 years ago and has replaced the metallic one. This is a commercially available pneumatic atomizing nozzle with a ceramic air cap (Reference 5-15, page 38). The ceramic unit is now used in all calcination operations and has shown minor wear (0.01 mm wear on the 6.3 mm nozzle orifice after 1000 operating hours). This remotely replaceable nozzle is expected to therefore remain in service for several months of operation (Reference 5-8, page 28).

The operation of the spray calciner is relatively simple with startup commencing as soon as the furnace (spray chamber) can be heated to the operating temperature. Shutdown requires discontinuation of the liquid waste feed and a one minute nozzle flush (Reference 5-8, page 28).

A major advantage of the spray calciner in relation to other solidification processes is that there is no reliance on physical contact of the waste on a hot surface. The incoming liquid waste concentration is not considered by Battelle to be of concern (Reference 5-20, pages 43-69). The drying of the waste is accomplished by a combination of radiative and convective heat transfer. Wall mounted pneumatic vibrators are periodically operated to prevent scale accumulation on the walls of the spray chamber (Reference 5-15, page 39).

5.2.1.4.3 Engineering Achievability

Remote Operation and Maintenance. Remote operation and maintenance are important to provide a safe non-radioactive environment for the facility personnel. It is also important to keep maintenance requirements as low as possible.

The components identified that can be expected to either wear out or fail are: the feed nozzle assembly, the off-gas filter assembly, and the vibrators (Reference 5-15, page 38). Consequently, there are provisions in the proposed BNWL full-scale remote facility (currently in design) for remotely replacing these components as well as the entire segmented spray furnace chamber. Figure 5-6 shows a number of the remote replacement items.

The spray calciner furnace connections for power and temperature indicator leads utilize connectors which facilitate remote operation and maintenance. The furnace is mounted on springs to reduce the vibration of the heating elements during the vibrator operation to remove scale buildup on the chamber walls. The vibration is set up by impact on a stainless steel plate attached to the furnace. If, and when, the furnace segments require removal, an in-cell crane is utilized (Reference 5-15, page 40).

The off-gas filter assembly is connected to the calciner by remotely removable flange bolts. Hanford-type connectors are used for remote disconnection ability of the jumpers that supply blowback air to the filters. The control valves for this blowback system are located outside the cell (Reference 5-15, page 38).

The vibrators can be remotely installed as they are mounted on Hanford-type connectors. In this case, the connectors serve a dual function by also providing the required air connection (Reference 5-15, page 38).

The process cell arrangement is shown in Figures 5-7 and 5-8. Arrangement considerations are given to operations, remote maintenance, solids flow, system hydraulics, and connecting piping routes (Reference 5-15, page 41). Operation considerations involve assuring adequate viewing and providing in-cell crane and manipulator access where required. Remote remaintenance provisions require that most major components are set for direct over-head crane removal. Gravity flow of calcine and glass frit is provided by equipment design and placement. To assist in decontamination of the piping interior and to assure that there is no liquid holdup or particle settling, all piping is sloped either to or from the point of origin (Reference 5-15, page 42). Also, removable jumpers are used to make all service and process connections.

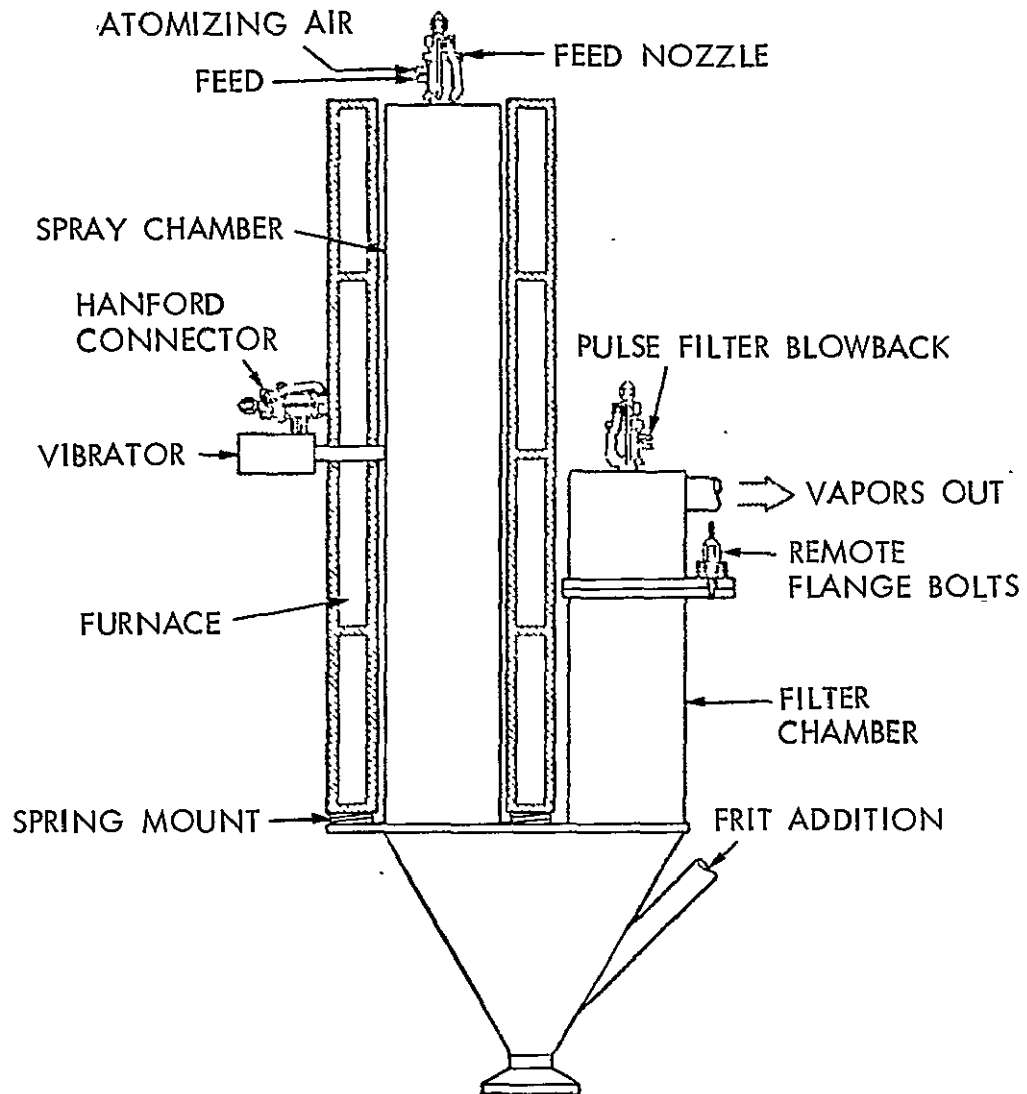


Figure 5-6. Remote Design Features of the Spray Calciner (Reference 5-15, page 39)

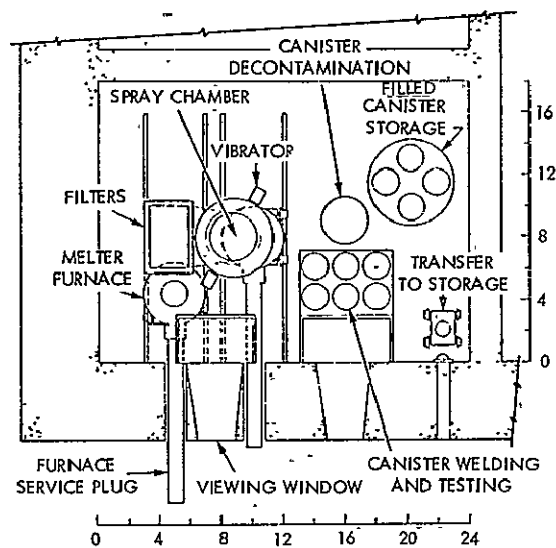


Figure 5-7. Cell Arrangement of Vitrification Equipment,
Plan View (Reference 5-15, page 41)

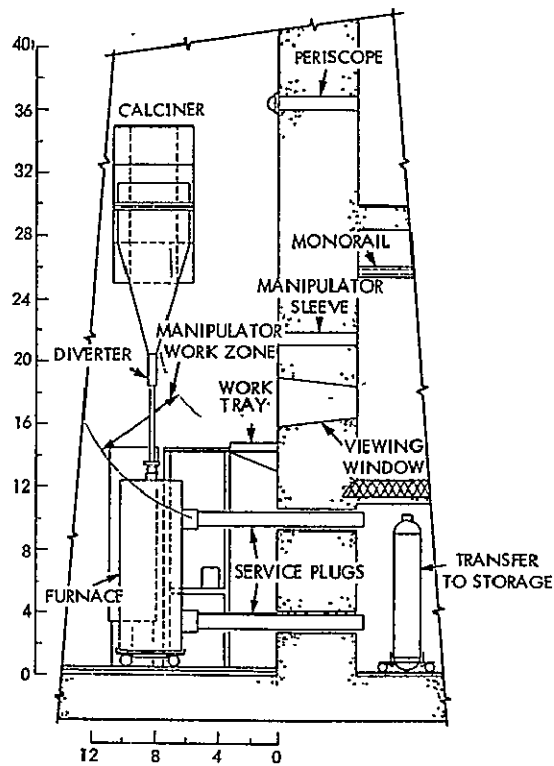


Figure 5-8. Cell Arrangement of Vitrification Equipment, Elevation View (Reference 5-15, page 41)

A full scale remote operation and maintenance non-radioactive facility is under construction to gain experience in remote cannister handling and maintenance techniques (Reference 5-9, page 20). Remote operation and maintenance pilot scale tests in a hot-cell facility (15 liters/hour) are to be carried out in 1978 (Reference 5-24, page 1).

5.2.2 Immobilization of Solidified Waste by Vitrification

The purpose of the incorporation of the calcined waste in a matrix material is to reduce the dispersability, volatility, and leachability of the waste solid. The characteristics of solidified wastes that are required to meet these goals include

- adequate thermal conductivity
- good thermal stability
- radiation resistance
- mechanical ruggedness
- minimal volume
- process reliability and simplicity.

Adequate thermal conductivity in the waste form is desired to prevent excessive internal temperatures during the initial few years of storage. A coherent ceramic waste form has a thermal conductivity that is 3 to 5 times higher than a loosely consolidated powdery oxide waste form.

The relative leachability of a particular waste form is a function of the surface area and specific reaction rate with water. It is important that neither radiation nor thermal effects deleteriously increase the surface area or leach rate at any time during storage or transportation. Also during transportation, because the canister is the first line of protection, it is important that the mechanical ruggedness of the canister not be weakened by the interactions between the waste and the canister.

With respect to plant operations, solidification should be done economically and the solidified waste product should occupy a low volume. Due to the fact that all solidification operations will take place in a shielded cell, process reliability and simplicity will be two major factors in minimizing costs. Also, a volume reduction factor of 7 to 10 which is generally achieved during solidification assists in minimizing the volume. For the mainline system in the United States as well as in France, the United Kingdom, Germany, and India, borosilicate glass has been selected to be the matrix material.

5.2.2.1 Process Description. The vitrification process developed for the U.S. mainline system is in-can melting (Reference 5-9, page 6). A schematic diagram of the in-can melter is shown in Figure 5-9. The storage

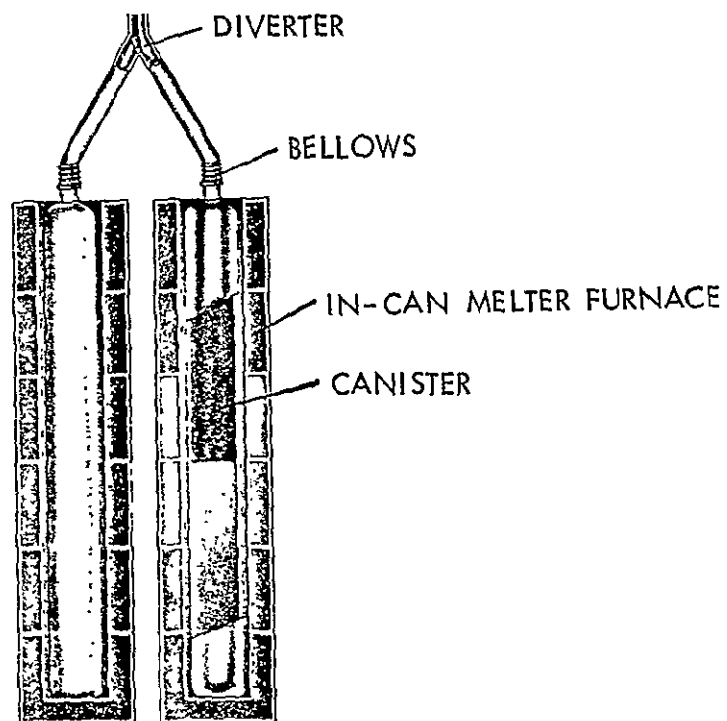


Figure 5-9. In-Can Melter (Reference 5-8, page 27)

canister is located in a multi-zone furnace and coupled directly to the calciner. Glass-forming frit is mixed with the calcined waste and the mixture is fused in the canister at 1000 to 1100°C. As the melt level rises, the lower furnace zones are turned off and cooling is initiated. The off-gas effluents are vented through the calciner offgas system. Once cooled, the canister is removed from the furnace, sealed, weld integrity checked, decontaminated, and inspected again prior to storage.

The glass-forming frit is metered continuously to the melter at a rate proportional to the calcined product generation. The flow rate is established from the known HLLW flow rate to the calciner and the known total oxide content of the HLLW. With the metering and hopper equipment located outside the hot cell the frit flow rate is easily maintained (Reference 5-11, page 6.41). The metered frit falls through an air lock and into a cone below the calciner. The potential for batch sintering in processing equipment due to the self-heating of the waste is minimized because the waste and frit are mixed just prior to entering the melter (Reference 5-8, pages 23-30).

Two melting furnaces are located below the calciner--calcining is a continuous operation and in-can melting is a batch process. A diverter valve is situated in the feed ducts to the furnaces. Once a canister is filled to the desired level, the calcine-frit mixture is diverted to the second melting furnace. The unfixed material on the inside surface of the canister above the final melt is melted in place (Reference 5-8, pages 23-30).

The canister is planned to be held in the furnace for several hours to assure complete melting, prior to being cooled, capped, and removed from the furnace. The cap is welded, the canister and weld are leak checked, decontaminated, and the integrity inspected before being sent to storage. (See Figure 5-11.)

The borosilicate waste/glass produced in the in-can melter is a vitrified, slightly fractured monolith which is sealed in a metallic storage canister. The processing restraint (relatively low glass forming temperature) of melting the waste in the storage canister prevents the complete assimilation of all the waste oxides in the glass and minor amounts of several different crystalline phases are dispersed in the glass (Reference 5-16, pages 1-8). The leachability of the glass has not been shown to be significantly increased by the presence of crystalline phases (Reference 5-20, pages 43-70).

5.2.2.2 Development History. Between 1966 and 1970 the Waste Solidification Engineering Prototypes program incorporated HLLW in glass utilizing various calcination and vitrification methods and four levels of simulated fuel exposure (Reference 5-10, pages 3.1-5.55). The WSEP equipment was scaled to handle HLLW from a 1 MTU/day reprocessing facility. In 1970, vitrification of HLLW using a spray calciner coupled to an in-can melter was operated at Battelle Northwest Laboratories during the WSEP program. Two fully radioactive engineering-scale runs were made (Reference BNWL-1583). Further development was initiated in

1974 in the Waste Fixation Program. Between October 1974 and 1975, 13 engineering-scale, nonradioactive in-can melting campaigns were carried out. The in-can melter was coupled to a spray calciner for several of the runs, with the remaining operations being carried out using a wiped-film evaporator and a fluid-bed calciner (Reference 5-10, page 3.3).

5.2.2.3 Program Status and Plans. The Waste Fixation Program (WFP) is continuing to develop the in-can melter coupled to a spray calciner. A full-scale non-radioactive facility (150 liters/hour) was completed and began operation in April 1977 (Reference 5-12, page 1). A full-scale non-radioactive remote operation and maintenance facility is in design to gain experience in remote canister handling and maintenance techniques (Reference 5-9, page 20). Radioactive tests will be carried out in a pilot plant-scale spray calciner/in-can melter process (15 liters/hour) in 1978 (Reference 5-12, page 1).

5.2.2.4 Issues Related to Technology Status of In-Can Vitrification

5.2.2.4.1 Scientific Feasibility

Glass Integrity. The effect of the radioactive, thermal, mechanical and chemical properties of the waste on the composite borosilicate glass ability to retain the radionuclides is the major item of interest under glass integrity.

Radiation Effects. The HLW-glass will be subjected to intense levels of alpha, beta, and gamma radiation as well as considerable levels of neutron radiation. After a period of storage, the major radiation effects will be due to alpha radiation and the associated alpha recoil nuclei. Beta and gamma radiation cause relatively minor amounts of damage. Because of their low fluxes in the waste, neutrons and fission recoils produce a minimal impact on the waste. The net rate of radiation-induced defects in the waste glass depends on both the defect formation rates and the defect recombination, or stabilization, rates. The alpha radiation induces the buildup of stored energy, density changes, mechanical structural changes, helium generation effects, and changes in leach rate (Reference 5-20, pages 71-80). The results of investigations of each of these phenomena are discussed below.

The buildup of stored energy in the solidified HLW is the result of displacement of atoms from their normal lattice position due predominantly to alpha recoil interactions. If a large quantity of stored energy is released in a short period of time, the result would be a sharp increase in the temperature of the glass. Such a temperature rise could prove to be detrimental to the integrity of the waste canister and the physical and chemical properties of the solidified waste-glass (Reference 5-18, page 1.1).

To investigate the buildup of stored energy, three glass types have been doped with Curium-244 for accelerated alpha radiation tests. The buildup of the stored energy in all three glasses (zinc-borosilicate, lead-borosilicate, and high silica glass) tended to be similar. The energy increased rapidly to 17-20 cal/gram and then began to level off.

No significant difference in energy buildup was observed in zinc-borosilicate glass using 1% and 8% curium-244 (for 3,000 and 50,000 year simulations). (Reference 5-22).

The amount of stored energy accumulated in glass decreases as the temperature is increased, decreasing nearly linearly with increasing storage temperature, reaching a negligible value at 350°C. The self-heating of actual wastes will therefore reduce the amount of stored energy, particularly during the first 50-100 years after vitrification.

The stored energy in zinc borosilicate glass will not likely exceed 50 cal/g. If this amount were released adiabatically, temperature rises in the glass would not be expected to exceed 200°C. Such a release of stored energy was determined to possibly result only from a rapid rise in the waste's temperature caused by an external heat source or lack of cooling source (Reference 5-18, pages 7.1-7.4). With the maximum release rate found to be 0.08 cal/°C gm there is expected to be no spontaneous or serious effect on the glass (Reference 5-18, pages 6.1-6.4). Such temperature rises, although significant, are not expected to have serious consequences (Reference 5-18).

Effect of Alpha Particle Radiation on Density of Glass. Small but measurable density changes (either positive or negative depending on the specific glass formulation) have been observed as a result of alpha particle radiation. Experimental results, using curium-doped (1%) zinc borosilicate glass, indicated a density increase (shrinkage) of about 0.2% for the devitrified glass and approximately 0.8% for vitrified glass samples (Reference 5-20, page 76). The significant influencing factor in these density changes does not appear to be radiation dose. For the was only a very small difference in density increase at the apparent maximum dose of 10^{18} alpha/gram (the density began to perceptively decrease at higher doses) with glass samples doped with 1% and 8% curium. The observed density increase of 1% for the 8% curium-doped zinc-borosilicate vitrified waste could possibly be large enough to affect the cannister design. (Reference 5-20, page 76). The affects of alpha radiation on density can apparently be controlled, if necessary, by varying the glass composition as indicated by the behavior of the three glass types studied (1% Cm-doped, vitreous-and devitrified-, and 8% cm-doped vitreous zinc-borosilicate glass). (Reference 5-20, page 77).

Effect of Contained Alpha Particle Radiation on Leachability of Glass. Glass leach rates have been determined by examining the behavior of curium and potassium in the glass samples used in accelerated alpha experiments. As shown in Figure 5-10, leach rates based on potassium are a hundred times greater than those based on cesium in the glass specimens. The irradiated and nonirradiated data for potassium based leach rates indicate there is only a slight increase in leachability of the reference glass due to high alpha doses. It was concluded by Mendel et al. that, since there was no significant difference in leach rates for samples with different doses, the doses used in these experiments produced radiation effects which exceeded the radiation saturation point of the glass (Reference 5-20, page 77).

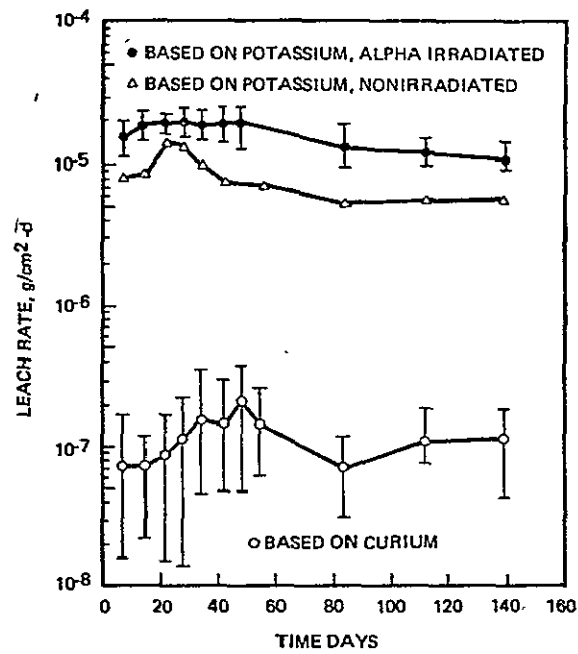


Figure 5-10. Leachability of ^{244}Cm -Doped Zinc Borosilicate Waste Glass (Reference 5.20, page 77)

Helium Behavior in Waste Glass. The helium generated from the alpha decay of the actinides must either be incorporated within the waste-glass structures, or be diffused to the surrounding atmosphere. Due to the waste-glass being contained in a sealed canister, there is a potential for helium pressurization and of the glass swelling because of the formation of helium bubbles. A detailed review of the experimental investigations of helium diffusion in curium doped zinc borosilicate glass is presented in Reference 5-21.

The glass samples examined contained $1-2 \times 10^{18}$ helium atoms/cm³ glass. Equivalent self-radiation damage and helium concentrations would be achieved after approximately 100 years of storage for glass containing commercial LWR generated waste. One of the major conclusions has been that in a canister with a 10 vol. % plenum, containing glass which is not a monolith, the pressure will not exceed 15 lbs/in² until the waste-glass has been stored around 10^4 years (Reference 5-20, page 79).

Other Radiation Effects. Mechanical strength, and visible surface changes are also being studied in the radiation effects task. Preliminary work to date indicates that no marked changes in these properties occur after alpha irradiation.

Thermal Effects.

Shock. During the cooling phase the waste-glass is expected to undergo thermal shock. The effect of this thermal shock on the reference reference zinc-borosilicate waste glass in a six-inch diameter canister is presented in Figure 5-11. As indicated in the figure, the surface area will increase by about a factor of 10 due to the thermal shock from air cooling.

These experiments indicate that with time, a secondary thermal cracking will occur for present waste disposal glass designs. For, as the major heat producers decay the thermal gradient in the canister will decrease. This decrease in gradient can produce sufficient stresses to crack the glass. Due to the fact that the temperature distribution within a cylinder of uniform heat generation is parabolic, the major stress is expected to occur near the canister walls. At the location near the walls the glass will already be cracked from thermal shock. Therefore, additional cracking from thermal stress will be limited (Reference 5.16, pages 1-8). What cracking that does occur will be a function of initial thermal power in the canister, the fin and canister design, the storage conditions, and the position of the glass within the canister. Also, the thermal shock sensitivity of a waste-glass will be dependent on the thermal conductivity, thermal expansion coefficient, and heat capacity of the glass--though variations in these properties have been minor for different waste glasses (Further investigations into this behavior of thermal cracking are planned for 1978).

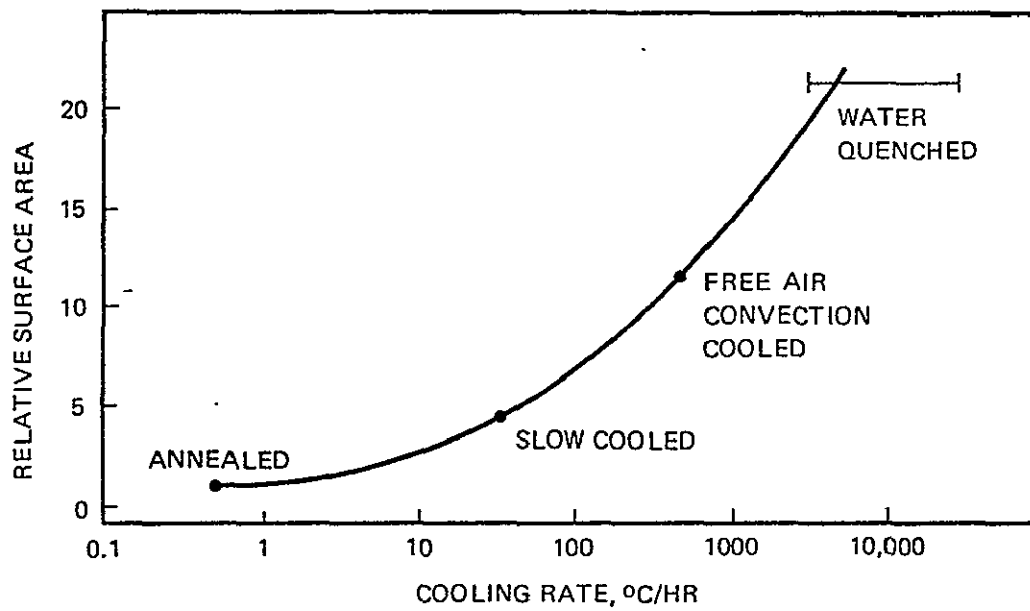


Figure 5-11. Effect of Cooling Rate on the Generation of Increased Surface Area in a 6-inch Diameter Glass Casting (Reference 5-20, page 36)

Devitrification. As a result of prolonged self-heating at certain temperatures devitrification may result, with corresponding increases in the leachability and porosity of the glass. Devitrification is the formation of crystals in glass. The rate of devitrification and the types of crystals formed is strongly dependent on formulation and thermal history but not significantly influenced by radiation (Reference 5-19, pages 19-25).

In the reference borosilicate nonradioactive glass samples, the maximum effects of devitrification occur between 700°C and 750°C (Reference 5-20, pages 33-91). The major devitrification products are zinc silicate crystals (Zn_2SiO_4) and a strontium molybdate phase. The zinc silicate crystals cause some minor cracking of the waste-glass because of a mismatch of the thermal expansion with that of the glass. The molybdate phase (either calcium, strontium, or barium) was found to have a greater water solubility than the waste glass. The reference glass properties are impacted to a smaller extent by the other phases present in the glass (Reference 5-20, page 57).

Long-term leach test results indicate that devitrification effects at 700°C increase the leachability of the reference glass by a factor of 10 (Reference 5-20, page 50). The devitrification changes the glass such that it "leaches" mainly by corrosion while attack of the vitreous glass is mainly by a true leaching mechanism.

Many complexities are involved in the leaching of vitrified and devitrified waste glass by very slowly moving water. A general indication of the durability of the reference zinc-borosilicate glass has been obtained by the extrapolation of the leach rate results for cesium and strontium by slowly moving 25°C water (Reference 5-20, pages 49-54). The depth of leaching over a period of 1000 years is estimated by Mendel et al. of Battelle to be equivalent to one millimeter of the original glass cylinder's diameter being leached from the devitrified exterior area of the glass cylinder. The vitreous interior glass is estimated to be penetrated less than a tenth of a millimeter in 1000 years (Reference 5-20, page 53). These estimates were based on cesium experimental data (Reference 5-20, page 54). For estimates beyond 1000 years strontium experimental data were used as the extrapolation base due to the fact that in actual high-level waste the cesium will have decayed away after 1000 years. The extrapolated results are interpreted by Mendel et al. to indicate that after 100,000 years the devitrified glass cylinder exterior could have been penetrated approximately one centimeter and the vitreous interior glass less than one millimeter (Reference 5-20, page 53). As stated previously, there are many complex factors involved in extrapolating experimental results that have been obtained over a relatively short period of time to long-term leaching predictions. Further experimentation and analysis is necessary before such extrapolations can be made with confidence.

Mechanical Shock Impact. Mechanical forces acting on a canister of solidified (e.g., vitrified) waste could not only breach the canister wall, but also break the glass into smaller, potentially respirable particles. The increased glass surface area would hasten

dissolution and volatilization if driving forces for these were present. Experiments have been performed (Reference 5-22) on the reference glass to estimate 1) the quantity of respirable glass fines produced; 2) the increase in glass surface area; and 3) the impact resistance of the filled canisters. Even if all test canisters were to remain intact, the first two items are needed because it cannot be guaranteed that every production canister will be fabricated soundly and maintained properly.

Tests were conducted using nonradioactive reference waste glass in cylindrical 304L stainless steel canisters. Both series included specimens which were essentially glassy and those which had been partially devitrified by thermal treatment. The canisters breached only at the two highest velocities (66 and 117 fps). The breaches were all very small cracks. Pre- and post-test weight checks indicated that very little, if any, glass escaped through the cracks. The inventory fraction of the resulting fines smaller than 10 microns typically ranged from 10^{-8} for control specimens to 10^{-4} for 80 mph impact. This compares with approximately 10^{-2} for nonimpacted calcined waste (and probably for impacted calcine also). Geometric calculations were made of the surface area created. The surface area typically increased by only a few percent of the initial surface area for control specimens, but by a factor of 40 for 80 mph impact.

No consistent difference was observed between the essentially glassy and the partially devitrified specimens nor between small and large canisters when the results were compared on a fractional breakup basis. Testing of specimens which are more severely devitrified might lead to observable differences between glassy and devitrified waste. Testing at elevated temperature increased the quantity of large particles produced, but no significant effect was observed on the quantity of particles smaller than about 20 to 50 microns.

In borosilicate waste-glass, two types of non-homogeneity have resulted: occlusion of some waste oxide near the bottom and on the top of the melt; and a separate microcrystalline phase which collects at the surface or as globules suspended within the melt (Reference 5-16, page 8). Though the occlusion appears not to affect product quality significantly, the separate phase is undesirable due to its being more water soluble and more corrosive than the bulk melt. It is also enriched in certain fission products -- cesium in particular. Either higher temperatures or increased duration of thermal treatment decreases the nonhomogeneity. According to McElroy of Battelle, these problems, which arose in the WSEP program, have been largely eliminated by changes in glass composition and thermal treatment (Reference 5-12, page 1).

5.2.2.4.2 Technical Feasibility

Glass Composition. The present program is concerned with several reference glasses and the factors affecting the composition.

There is not one specific type of waste glass suited to the immobilization of all LWR high-level wastes. Waste glass compositions must be adapted to account for many variables, including (Reference 5-20, page 1):

- Reprocessing plant operating mode - which affects the amount of U and Pu in HLLW
- Frequency of plant shutdowns - which affects the amount of sodium and other inerts in the waste
- Waste vitrification - which affects restraints placed on allowable glass viscosity, electrical conductivity, redox potential, etc.
- Fuel type and burnup - which affects the ratio of fission product to inerts in HLLW, the amount of transplutoniums, and possible presence of fluoride
- Reprocessing plant flowsheet - which affects the amount and type of inerts in HLLW (e.g. presence or absence of soluble poisons, ILLW, Pu reductants).

Glass is created when a mixture of glass formers, intermediates, and modifiers are heated. The glass formers are those oxides with high bond strength (>80 kcal/mole) while intermediates have bond strengths between 60 and 80 kcal/mole and modifiers are any oxides that fall below 60 kcal/mole (Reference 5-9, page 3).

One or more of the glass formers (SiO_2 , B_2O_3 , and P_2O_5) are essential for any waste glass. The characteristics of the glass are determined by the intermediates and modifiers. The intermediates (such as Ti and Al) generally tend to increase the melting point as well as increase the durability. The modifiers (including alkalis and alkaline earths) aid in melting but can decrease durability. With the exception of some phosphate in the waste, the HLW constituents act as modifiers and/or intermediates. Supplemental intermediates and modifiers are then added to produce the desired waste-glass characteristics (Reference 5-9, page 4). The general compositional range for borosilicate waste glasses is shown on the ternary diagram in Figure 5-12.

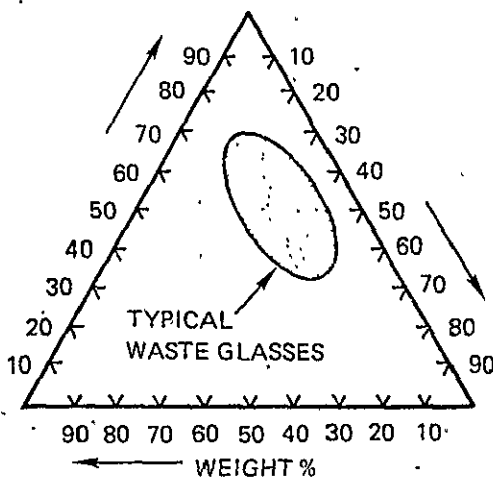
The properties of the waste-glass are determined by composition, processing techniques, and handling. The current Waste Fixation Program has examined in considerable detail a reference glass composition with 23 wt% radioactive waste (Reference 5-8, page 12). This generic investigation is being undertaken so a detailed definition of the behavior of one characteristic glass formulation can be extrapolated to other formulations within the same class with a less extensive number of experiments. In addition, investigations have begun on three other waste glass compositions representative of past or planned fuel reprocessing plants. These are the formerly

RADIOACTIVE
COMPONENTS
OF WASTE

FISSION PRODUCT OXIDE
TO ACTINIDE OXIDE RATIO
MAY VARY FROM ~ 3
TO MORE THAN 15

GLASS FORMERS

$\text{SiO}_2\text{B}_2\text{O}_3$ RATIO MAY VARY
FROM ~ 1.3 - 6. PHOSPHATE (FROM
SOLVENT DEGRADATION) USUALLY
NOT MORE THAN 10% OF GLASS
FORMERS



INERTS

DERIVE BOTH FROM WASTE
AND GLASS FRIT. WASTE
CONTRIBUTES Na_2O , Fe_2O_3 ,
 Cr_2O_3 , NiO , POSSIBLY Gd_2O_3
AND OTHERS. FRIT
CONTRIBUTES MAINLY M_2O
AND MO

Figure 5-12. General Range of Composition of Borosilicate Waste Glasses (Reference 5-20, page 2)

operational Nuclear Fuel Service Plant in West Valley, New York, and the proposed Allied General Nucleus Services (AGNS Burnwell, S.C.) and Exxon plants. These formulations will consider 23 wt% radioactive waste.

In general an empirical approach to establishing guidelines for glass formulation has been made (Reference 5-9, page 12). The initial testing guidelines are meltability (2-3 hours at 1050°C) and leachability. Meltability is improved by certain composition additions while leachability is often increased by these additions (Reference 5-9, pages 12-13). Waste oxides can be incorporated in all of the glasses investigated to a level of between 40 to 50 wt%. But 23 wt% has been determined to be the optimum amount by McElroy et al. of Battelle based on considering processing factors (volatility, homogeneity, corrosion), leachability and economic factors (Reference 5-8, page 12). Reduced quantities of waste could be incorporated in the glass if regulations required such a step -- although according to McElroy the glass composition, that has been developed and studied, would not be suitable for a waste content of less than 16 wt% (Reference 5-12, page 2).

5.2.2.4.3 Engineering Achievability

Remote Operation and Maintenance. The potential difficulty of remote maintenance and the need for safe remote operations are areas of significant concern in the design and operation of the vitrification process.

The in-can melter furnace in the proposed BNWL full-scale remote facility design is mounted on a structure that has service plugs extending through the cell-wall to allow for external connecting of all service lines. The furnace structure is set on rails to allow movement of the canister to a position under the spray calciner. The load cells (Figure 5-13) and furnace retort can be replaced remotely (Reference 5-15, pages 40-42).

It may be required that the furnace retort have an inert atmosphere during melting to reduce spall (scaling - oxidized surface coating on stainless steel). This will be accomplished by a seal connected to the coupling assembly (Figure 5-13). The coupling assembly uses air couplers and metal bellows. The bellows allow for misalignment, thermal expansion, and provide vertical movement necessary during canister changing.

Remote maintenance of the two in-can melters is obtained by an in-cell crane once the melters are rolled to the back of the cell. At this location the service plugs will have cleared the wall. Shielded viewing windows and/or periscopes will be provided as required (Reference 5-15, page 40).

With respect to the frit addition equipment, all failure components will be located outside the cell. An air break and airlock to prevent air inleakage to the calciner and backflow of contaminated material are located in a shielded wall niche. All piping is sloping to assure free drainage and to assist in decontamination.

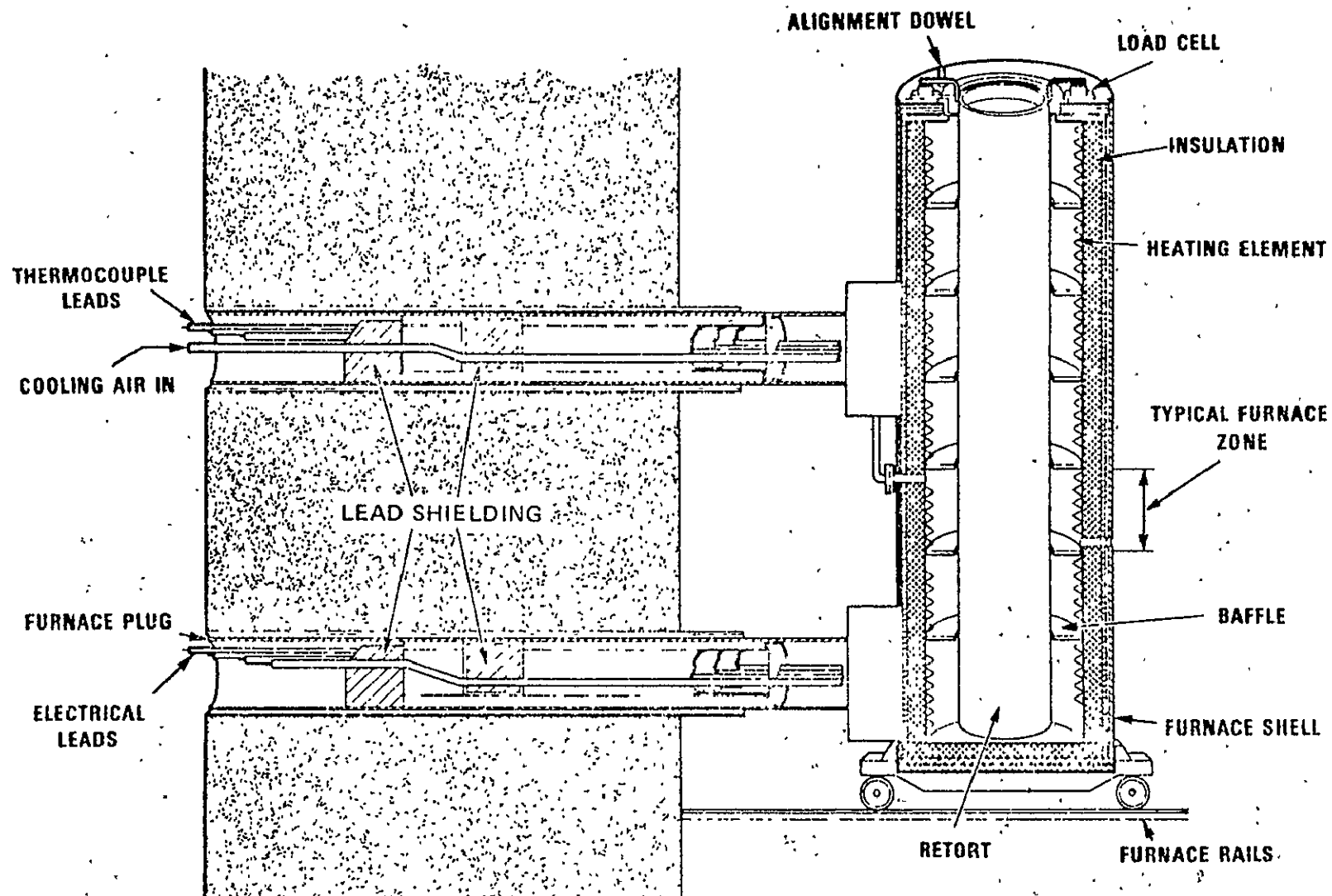


Figure 5-13. Multizone Resistance Heated-Air Cooled Melter Furnace (Reference 5-15, page 40)

Verification tests of selected remote features is currently underway at BNWL on a "plant size" (150 liters/hour) scale. A full-scale non-radioactive remote operation and maintenance cell will then be constructed. Also, pilot scale (15 liters/hour) remote demonstration tests will be undertaken in a hot-cell using commercial high-level waste in 1978.

5.2.3 Encapsulation

5.2.3.1 Process Description. The function of the canister is to provide primary containment of the waste during its surface storage and possible retrieval period in the geological formation. The main criteria for the canister is that it has a strength and corrosion resistance over a range of temperatures and retain a leak-proof seal.

The canister will be capped and removed from the furnace after a sufficient cooling period for the melt. The cap will be welded, checked, and the canister decontaminated prior to transfer to temporary storage. A schematic diagram of this main process is included in Figure 5-14.

The option of overpacking (placing the primary canister in a secondary canister) has been investigated and may be required by future Federal canister criteria for repository storage. Such criteria requiring overpacking could involve reducing surface contamination levels or the handling of the primary canister as a pressure vessel (Reference 5-11, pages 6.8-6.9).

The simplified decontamination of the secondary canister and the improved quality of external containment would be realized at the cost of reducing the heat content of the primary canister and increasing the waste containment expense (Reference 5-11, page 6.9).

5.2.3.2 Development History. The canisters used in the WSEP program were from 6 to 12 inches in diameter and 8 feet long (Reference 5-10, pages 7.1-7.4). The current typical canister design in the WFP program is a cylindrical canister 14 inches (36 cm) in diameter and 10 feet (3 m) long containing 245 245 liters of glass (Reference 5-9, pages 7-11). This canister also has an internal fin arrangement which increases heat transfer into the canister during melting and later assist in the removal of heat from the waste-glass mixture. The fins can allow the almost doubling of the canister diameter with the same center-line temperature over a non finned design (Reference 5-9, page 7).

5.2.3.3 Program Status and Plans. The Federal repository criteria on the canister design have not yet been finalized. Some of the probable limits are considered to be (Reference 5-11, page 6.8):

- maximum diameter, 40 to 60 cm
- maximum length, 3 to 4.5 m
- maximum heat density, 2.9 kW/linear meter

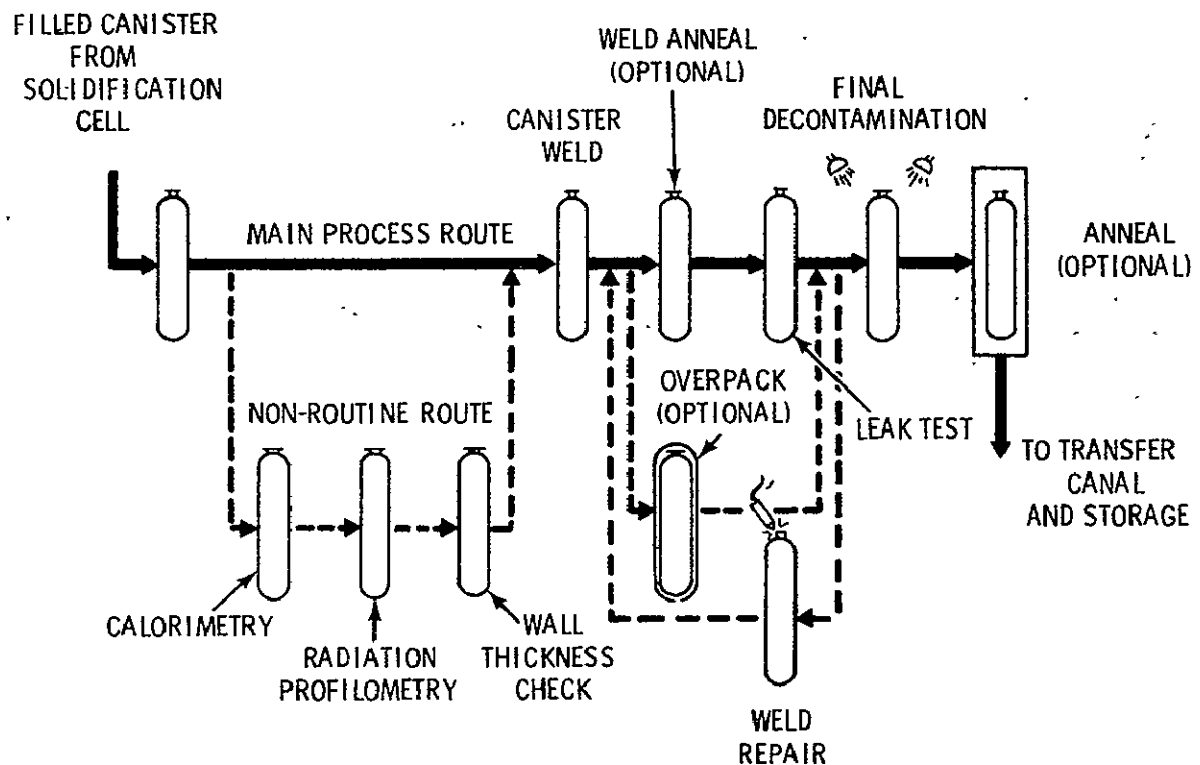


Figure 5-14. Typical Post-fill Treatment of Solidified HLW Canisters (Reference 5-11, page 6.9)

- maximum radiation, 1×10^{-3} rem/hr neutron dose rate and 1×10^6 rem/hr gamma dose rate, measured 1 m from any canister surface point
- surface decontamination low enough to maintain acceptable contamination levels at the repository

These limits will apply only after the solidified waste leaves the solidification facility. Heat, radiation, and density limits will be exceeded when the canister is being filled. At the time of storage, the heat removal capabilities, and allowable centerline and surface temperature are the limits.

Studies are continuing on container design and the finalization of the design is considered one of the major development program objectives of the solidification-immobilization program.

5.2.3.4 Issues Related to Technology Status of Encapsulation.

5.2.3.4.1 Scientific Feasibility. Canister materials factors are an important consideration in the establishment of scientific feasibility of the encapsulation process.

Those material factors that are significant during the melting operation include internal and external corrosion and potential deformation as a result of metal creep (Reference 5-9, page 8). Although no significant canister deformation has occurred in the pilot plant runs utilizing 304L stainless steel canisters, the molten glass level never rose above 4 feet because of furnace limitations. A deeper furnace has been acquired (April 1977) in order to study the effects of filling the longer canisters with a full load of glass (Reference 5-24, page 1). Early results with 12 and 16-inch diameter canisters with over 6 feet of glass are giving results similar to the 4 foot fill data. (Reference 5-24, page 1). Stress rupture tests have begun in order to obtain data on the chrome nickel steel canister candidate materials for temperatures up to 1000°C .

One of the two types of external corrosion is spalling (scaling) of the oxide scale from the canister surface. The scale can create crevices for points of corrosion during water basin storage. The scale also falls from the side of the canisters to the bottom of the furnace creating a minor secondary waste (Reference 5-9, pages 8-9). To reduce scale formation, inert gas purging of the furnace is being considered (Reference 5-15, page 40). The second external corrosion problem, sensitization, (a time-temperature process occurring at $400-900^{\circ}\text{C}$) can be minimized with proper cooling (Reference 5-9, page 8). During handling and storage the prolonged temperature of the surface of the canister should remain below 400°C to prevent sensitization. The internal corrosion situation during melt is not considered a significant problem -- occurring at a rate of 0.1 inch/month at 1050°C (Reference 5-9, pages 7-11).

During most of the handling operations the canister will be air cooled resulting in some temperature fluctuations. Decontamination (with spray steam or water) of the canister and canister emplacement in a water

basin will cause rapid temperature changes. The removal from the water basin for transport to the Federal repository will result in a temperature rise. The results of the temperature fluctuations will be some fracturing of the glass and a stressed condition on the canister. The stressed condition will be due to the lower thermal expansion of the glass compared to the canister (Reference 5-9, page 9). Failures of the canister resulting from the stress corrosion can be prevented if the water's chloride content is less than 1 ppm and the pH is alkaline (Reference 5-9, page 9).

Other materials being examined besides the current 304L stainless steel include 310 and 347 stainless steel, Inconel alloy 600 and 601, and Incoloy alloy 800 and 802. If "overpacking" is required, mild steel or cast iron could be used for the secondary container (Reference 5-9, page 9).

5.2.3.4.2 Technical Feasibility. The major issue in an acceptable technical process is the containment criteria the canister must meet. The current requirements (10 CFR 50, Appendix F) for containment specify that the canister may be held for a maximum of five years at the reprocessing facility and must retain its integrity for 90 days after emplacement at the repository. As discussed in the Program Status and Plans section, Federal repository criteria on the canister design are being developed to specify the specific requirements the canister must meet once it leaves the solidification (reprocessing) facility.

Process limitations on the canister dimensions result mainly from the considerations of high temperature materials strength and heat removal. Internal fins were included in the design in order to improve the transfer of heat into the glass during processing and out of the glass during storage.

5.2.3.4.3 Engineering Achievability

Long-Term Canister Integrity. With a possible requirement for retrievability of the waste glass canister from the geological repository over some extended period of time, the long-term canister integrity becomes of some concern.

During the WSEP program (1966-1970) 33 canisters were filled with various waste-glass composition and placed in either water basin, air basin, or ambient air storage (Reference 5-10, page 3.3). Of the three materials used in the WSEP canister construction, only 304L and 310 stainless steel were suitable for the in-can melting process (26 canisters). The canisters are still in storage at BNWL (Reference 5-9, pages 9-11), under varying conditions. The three canisters holding borosilicate glass were examined after about 2.5 years of storage with no observable changes in the product quality due to radiation decay (Reference 5-9, pages 9-11). One of the major results of these examinations is that the outside surface becomes sensitized during prolonged air storage at elevated temperatures (ambient air storage). It was concluded by Battelle that to avoid canister failure due to sensitizing, such ambiently-stored canisters should not be subsequently stored in water basins (Reference 5-13, pages 40-46). This and other results concerning interim storage integrity obtained from the WSEP program have provided a basis upon which to prevent such canister deterioration during interim

storage and thereby assist in prolonging the canisters' retainment abilities in the geological repository.

The current WFP program includes the examination of canister corrosion under both processing and long-term storage conditions (Reference 5-13). The actual canister material and design will be dependent on the Federal repository requirements. A number of materials and design configurations are being examined. It is expected that, from these various designs and materials, a canister can be developed that would provide the necessary long-term containment for possible retrievability to meet the presently non-finalized Federal repository requirements.

5.3 DEEP GEOLOGIC ISOLATION IN BEDDED SALT

5.3.1 Reference Repository Description. Figure 5-15 shows a cut-away of a conceptual repository prepared by the Office of Waste Isolation (OWI) which is managing the National Waste Terminal Storage (NWTs) Program for DOE. This repository, in concept, consists of surface facilities; shafts for waste, men, and material; and ventilation for a large number of excavated rooms at depths between 800 feet and approximately 3000 feet below the surface (Reference 5-25).

The reference repository will have an excavation underground that will cover approximately 2,000 acres. DOE will acquire surface and subsurface title to this land. On the surface, the waste receiving and support buildings will occupy approximately 200 acres. This will be the only visible evidence of the repository.* The remainder of the 2,000 acres may be leased for selective usage. Surrounding the 2,000 acre area overlying the excavation will be a controlled area for which ERDA will acquire subsurface title. This would permit all deep mining and drilling operations to be controlled, with the intent of eliminating any compromise of the safety and integrity of the disposal areas (Reference 5-25).

In the OWI concept, there are four shafts servicing the repository; a ventilation shaft, a man-and-materials shaft, a low-level radioactive waste shaft, and a remote waste handling shaft. The waste would be delivered to the surface facilities of the repository in a solidified form in specially designed containers. Before being lowered to the disposal area through vertical shafts joining the surface to the mine, each container will be examined for damage and surface contamination and an further overpack shield will be provided, if necessary.

High-level remotely handled waste will be unloaded from the delivering vehicle and lowered through the shaft to the disposal area, where it will be received by a shielded transporter vehicle. It will then be moved into one of the disposal rooms, where it will

*During the construction phase the mined rock and salt would be transported away from the site by truck and/or rail, as in conventional large scale underground construction projects. Decisions about the eventual disposition of the mined salt have not yet been made.

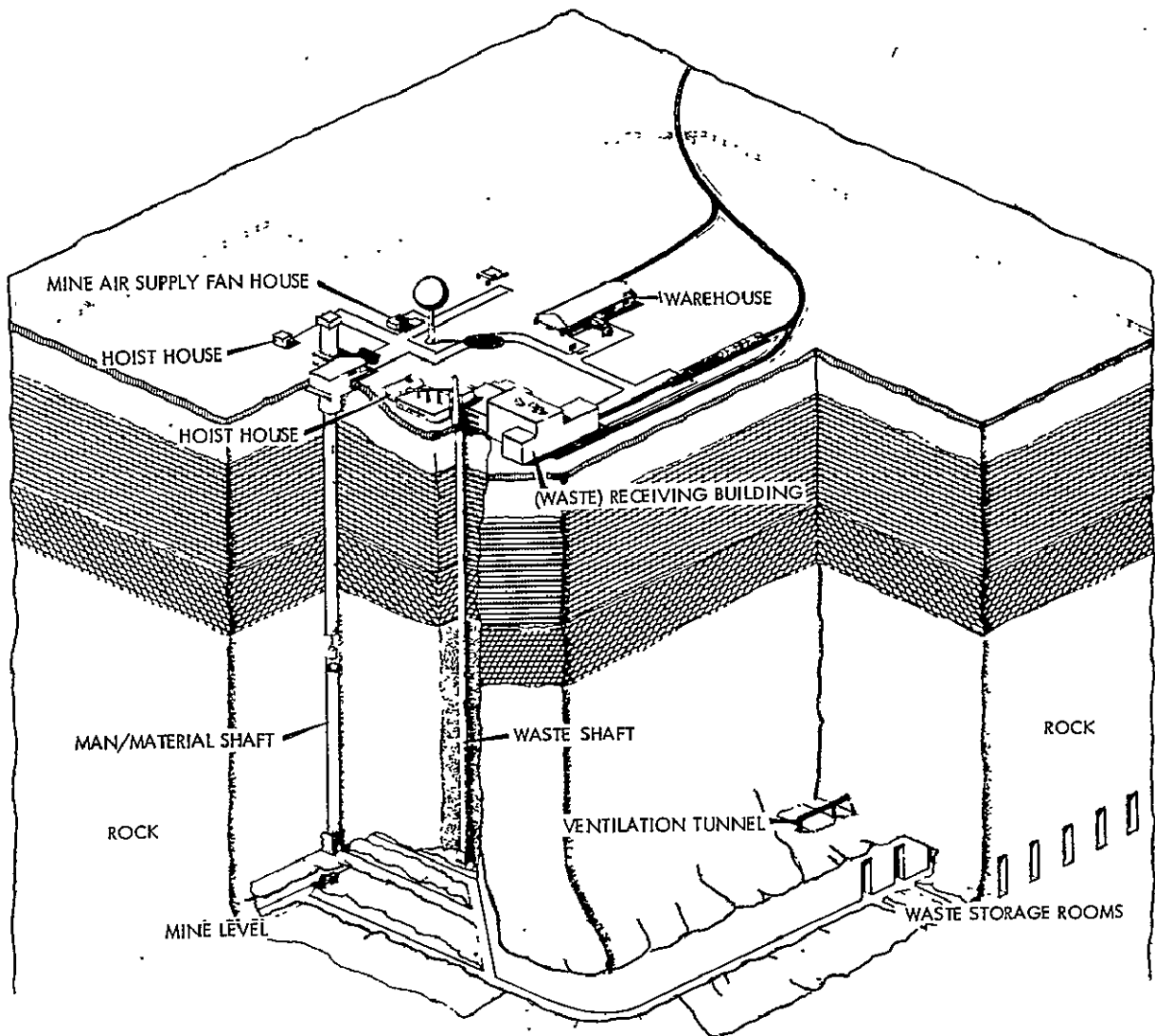


Figure 5-15. Simplified Cutaway of Repository
(Reference 5-20)

be lowered into a hole in the floor of the room, and the hole will be plugged with a shield for radiation protection. In the case of low-level waste, the containers will be lowered through the shaft and then stacked in the disposal rooms.

For the closed nuclear fuel cycle, all of the waste expected to be produced through the year 2006, based on the scenarios of 510,000 MW(e) production by the year 2000, could be put in one repository. This would require a repository to be designed to hold 6,000 cubic meters of high-level waste, 10,000 cubic meters of cladding waste, 45,000 cubic meters of intermediate-level waste, and 136,000 cubic meters of low-level waste. Of the total repository area, 70 percent is for high-level waste, 25 percent for cladding and intermediate-level waste, and five percent for low-level waste. (See Section 6.4.)

The mine layout is a conventional room-and-pillar layout. All storage rooms that receive canisters will be designed to have only one entry. This provides positive control for heat, ventilation, and exhaust for each room. Rooms for low-level waste storage will be designed as an open grid system to permit easy storage.

In the high-level waste disposal area, 32 containers will be stored in each room in vertical holes bored in the centerline of each room on 17-foot centers. Each room is about 150 feet long (Reference 5-25). Ceiling heights of 18 feet may be required to allow equipment operation and placement of waste canisters.

The waste will be retained in a retrievable mode so it can be removed from the repository with relative ease should circumstances develop that require this action. After the repository has operated for a period of time (10-50 years) and if no difficulties have occurred, the retrievable mode will be abandoned once the repository is full, and the rooms will then be backfilled with salt and sealed. The surface facilities will be decommissioned and dismantled, and all shafts will be plugged and sealed. The location will then be permanently marked.

5.3.2 Development History. In 1955, a conference was held under the auspices of the National Academy of Sciences (NAS) to discuss the possibilities of disposal of high-level wastes in geologic formations. This conference provided the first comprehensive discussion of deep geologic salt formations for waste disposal. In 1957, a report by the NAS Committee on Waste Management recommended salt as the most promising method of disposal (Reference 5-26, Appendix D).

With this impetus, Operation Salt Vault and other projects were initiated by the Oak Ridge National Laboratory (ORNL). The experiments in Operation Salt Vault were designed to determine the consequence of exposing bulk salt to radiation and heat. An abandoned salt mine near Lyons, Kansas was the chosen site. The experimental findings in the Salt Vault project on thermal response of the salt to embedded heat sources, thermal stresses in mine pillars, and emplacement and handling of spent fuel have been overshadowed by the subsequent failure of the Lyons, Kansas site as a possible repository. The repository project

in Kansas was officially terminated in February 1972 (Reference 5-27). But this failure is unrelated to the experimental findings. The experimental results obtained in Salt Vault provided important verification of the analytical heat transfer models and design concepts for a bedded salt repository, at least in the "near-field" of the disposal layer.

5.3.3 Program Status and Plans. The National Waste Terminal Storage (NWTS) program was established in 1976 to locate waste storage sites and to develop and manage Federal repositories. The general plan calls for the study of three types of geologic formations in 36 of the continental states and the eventual construction of six radioactive waste repositories. Locations of known large salt deposits in the U.S. are shown in Figure 5-16. With the plan for studying salt formations and other geologic formations, the number of site studies, and the number of planned repositories, ERDA expects the following advantages:

- (1) A greater probability of timely operation of a terminal storage facility is achieved due to simultaneous and parallel activities;
- (2) Retrievability of waste is accommodated because other repositories are available to receive the waste should it be necessary to move it for any reason;
- (3) It will not be necessary for one site or location to serve the entire country;
- (4) Waste transportation costs may be reduced if more than one facility is used since they will be dispersed around the country; and
- (5) The concern about Governmental reluctance to abandon a proposed site after significant expenditure is alleviated because other sites or repositories will be available.

The overall NWTS program calls for the establishment of six repositories; the schedule for which this work is envisioned is that the initial phase of operation of the first repository is planned to start late in 1985. The remaining repositories are also scheduled on about a 2-year phasing plan. A more detailed schedule for the design and construction of a repository is shown in Figure 5-17 (Reference 5-25). Although no sites have been selected, it is anticipated that Repositories One and Two will be established in salt formations (Reference 5-11).

The mainline system is that subset of the NWTS program that deals with the establishment of two commercial waste terminal storage facilities in bedded salt deposits that are scheduled to begin pilot plant operation in 1985. A concurrent system of similar intent, the Waste Isolation Pilot Plant (WIPP), has been set up to dispose of wastes generated in the defense program (Reference 5-28). The WIPP site has been selected; the repository will be constructed in a bedded salt deposit, and is slated to begin operation in 1983. A time table is presented in Figure 5-18 (Reference 5-28).

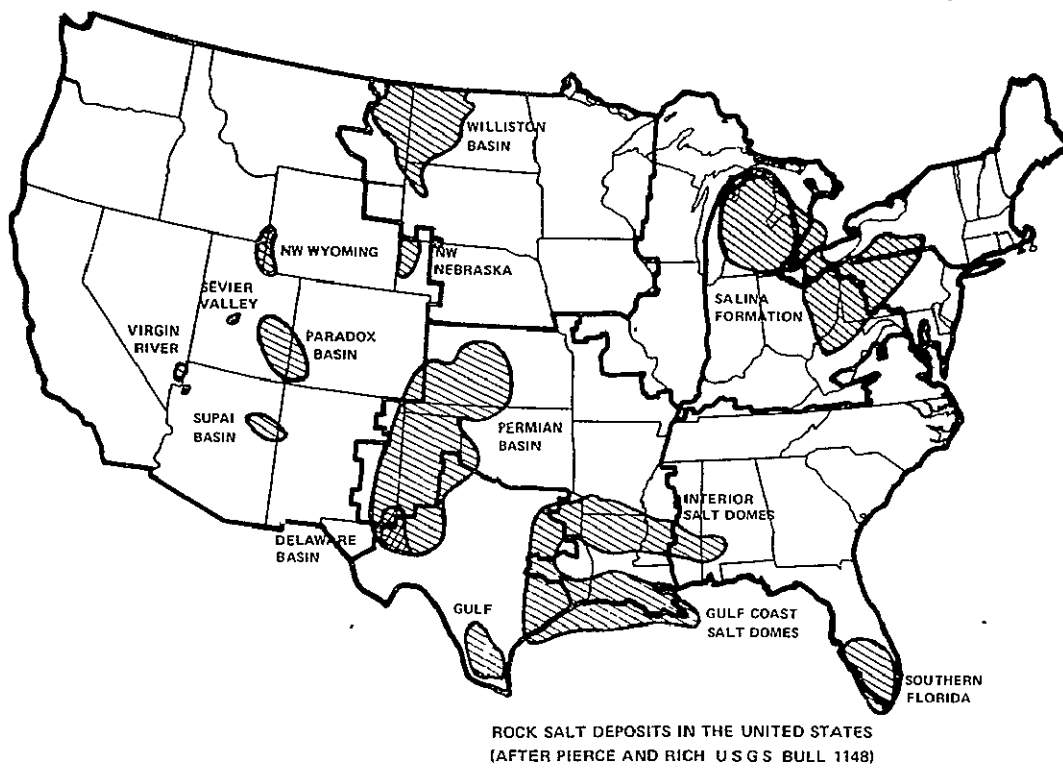


Figure 5-16. Rock Salt Deposits in the United States
(Reference 5-20)

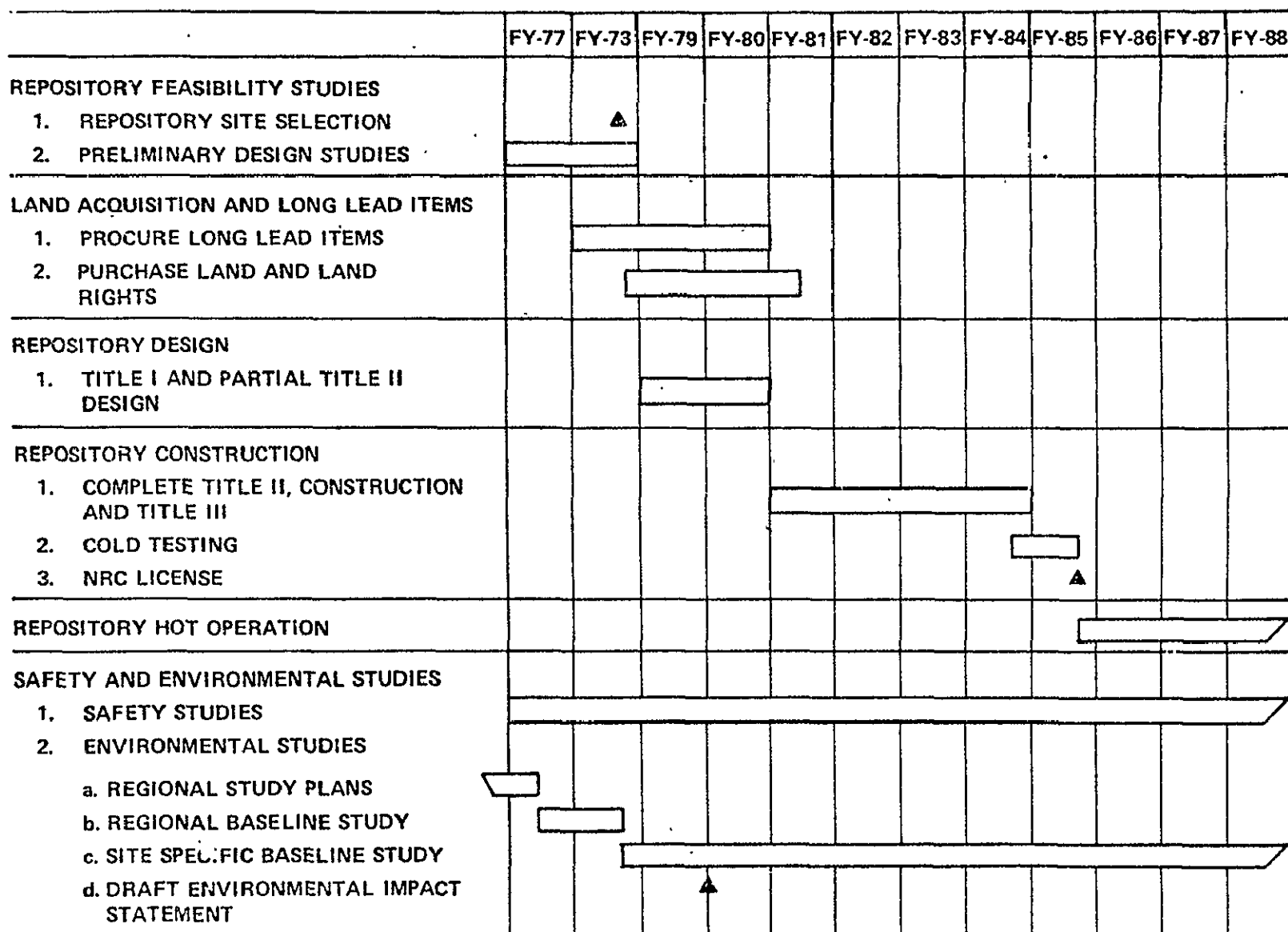


Figure 5-17. Geologic Terminal Storage, Repository No. 1 Design and Construction Schedule (Reference 5-25)

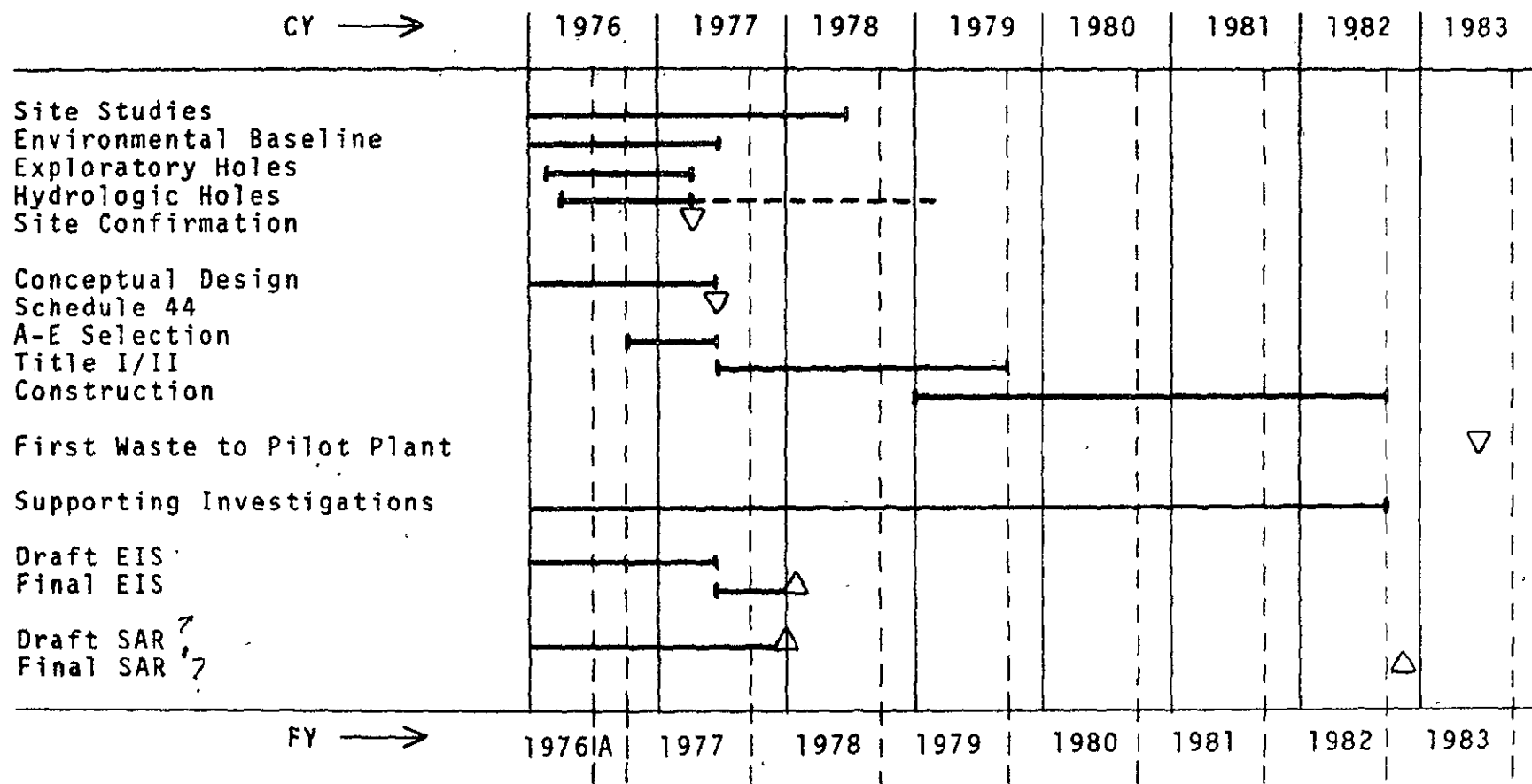


Figure 5-18. WIPP Milestone Chart (Reference 5-28)

5.3.4 Issues Related to Technology Status of Bedded Salt Isolation. The purpose of this section of the report is to examine the status of technology development for the use of bedded salt deposits as storage and disposal sites for high level solidified waste.

There is one obvious reason why bedded salt deposits have been the subject of long standing interest as sites for storage and disposal of radioactive wastes: the very existence of vertically thick and laterally extensive beds of highly water-soluble evaporite salts that have endured through long periods of geologic time demonstrates that either (1) the interiors of these deposits have never been exposed to circulating ground water; or (2) if they have been briefly exposed to circulating ground water the cracks that allowed ingress of water have apparently healed after formation.

The interiors of salt deposits, then, represent natural nearsurface systems that have been apparently stable for long periods of time ($>10^6$ years). Since circulating ground water would be the primary mode of transport of radioactive wastes to the surface of the earth where they can become entrained in the biosphere, salt deposits represent an attractive site for long term isolation.

5.3.4.1. Scientific and Technical Feasibility. The ultimate criterion for the establishment of repositories is the ability to guarantee long term isolation from the biosphere of the disposed waste. Any such guarantee for repositories established in bedded salt deposits relies upon the ability to predict:

- (1) the long term natural stability of any particular site, including seismic stability,
- (2) changes in the long term stability of a given site caused by mining operations during the establishment of a repository,
- (3) effects on the long term stability caused by the presence of radioactive wastes, and
- (4) the possible modes of breaching primary isolation and resultant transfer of materials from the disposal site.

Evaluation of the present or expected adequacy of the knowledge base for making these predictions requires a close examination of the following aspects of the data base:

- (1) thorough and lengthy geological, hydrological, geophysical, and seismic risk studies of individual sites, chosen for specific examination after regional surveys;
- (2) salt mining and emplacement/retrievability technology and expertise; and bore hole plugging and shaft sealing technology;

- (3) the results of two R&D projects involving the storage of radioactive materials in abandoned salt mines;
- (4) experimental theoretical, and field work on various problems related to chemical, thermal, and mechanical effects on salt caused by the presence of radioactive and thermally hot materials; and
- (5) theoretical, experimental, and field work on the release and transport of radionuclides by ground water intrusion.

5.3.4.1.1 Geology and Site Criteria Development. Site selection is the end result of a nested set of studies and experiments that the OWI has detailed as:

- (1) Identification of formations of interest
- (2) Reconnaissance surveys
- (3) Area studies
- (4) Detailed confirmation studies
- (5) In situ tests (Reference 5-25)

These studies are not necessarily carried out in a linear sequence. The intent of these studies is twofold:

- (1) to locate sites that satisfy general screening criteria, and
- (2) to acquire the knowledge necessary to predict the long term stability of a given site

General screening criteria are easy to establish and are discussed in a number of reviews (Reference 5-11, Volume 5). The information necessary to predict stability for any given potential site must come out of an integrated set of geological and hydrological studies whose objectives are tailored to each unique environment.

The OWI has the responsibility for commissioning and evaluating these studies. Their area coverage and progress towards site selection is reviewed in Reference 5-25. None of their studies have progressed beyond step (3) of the procedural outline above.

The weakest links in site evaluation and selection process seem to be:

- (1) specification of hydrogeological criteria necessary to ensure site stability;

- (2) understanding, measurement, and modelling of the current ground water regime in any given area of interest; and
- (3) extrapolation of that ground water regime up to one million years into the future.
- (4) understanding and prediction of transport of radio-nuclides should groundwater intrusion occur. (See Section 6.4)

There is no reason to anticipate that existing, planned, and recommended work will not result in a scientific data base which will permit site selection, although the anticipated schedule may be too ambitious. Each of the three sites which have come under the most intensive investigation has serious problems that has or could lead to its rejection as a repository. However, the extensive information gained, and the extensive data base that has been obtained could lead to better evaluation of new sites.

The Lyons, Kansas, Project Salt Vault site was an existing mine selected initially for evaluation and for R&D on the concept. It was later proposed as an actual repository for storage and isolation of fission products up to 700 years but was rejected partially because of hydrological problems encountered in a nearby mine, and because the abundance of drill holes (man-made hydraulic connections) in the surrounding (commercially exploited) area (Reference 5-29) decreased confidence that the site could meet the geologic and geographic isolation requirements.

In West Germany, the Asse salt mine near Karlsruhe was selected as an existing mine with a body of geological and stratigraphic knowledge available, which would be suitable for an R&D site for the storage of radioactive waste. Upon evaluation there appeared an incompletely understood but potentially serious hydrological problem. Specifically, there was earlier flooding in nearly abandoned salt mines, and there exists standing brine in the mine (Reference 5-30). This mine was rejected as a possible site for high-level waste, and it is used solely for low- and intermediate-level wastes to gain experience in waste disposal operation.

The site-specific nature of the geological location required is underscored by the experience at a proposed WIPP site in New Mexico. Drilling at the WIPP site in Carlsbad, New Mexico, turned up unexpected complexities in subsurface geology - features that are now understood, but which resulted in a shift in location of the proposed facility by about 7 miles (Reference 5-31).

There exist at present no sites where one can begin construction of a repository with the assurance of no future setbacks. However, the location of sites that will meet all criteria, both developed and as yet undeveloped, can be expected as the end product of the OWI site selection activity although the OWI schedule that calls for beginning cold testing operations by 1985 may require revision.

Any OWI work must also be integrated with the additional studies and experimentation being conducted by Nuclear Regulatory Commission (NRC), Environmental Protection Agency (EPA) and their contractors, such as the Lawrence Livermore Laboratories (LLL) studies. The LLL studies will develop performance criteria for solid waste and site suitability needed for the NWTs program. The NWTs program also intends to develop necessary and sufficient design criteria. The acceptability of the criteria will be determined by numerous reviewers. Obviously legislative and judicial hearings as well as public hearings and pending legislation will impact the development of criteria, and the development of technology to meet the criteria; as well as the scheduling of the activities.

5.3.4.1.2 Salt Mining and Emplacement/Retrievability Technology. Conventional salt mining techniques and technology are adequate to design and mine caverns in bedded salt deposits since (1) a repository is designed along the lines of a conventional salt mine; and (2) abandoned salt mines have been used with satisfactory performance for the retrievable storage of radioactive materials at both Lyons, Kansas and Asse, West Germany. Reviews of state-of-the-art mining technology are being carried out by both OWI and Sandia Laboratories.

Backfilling of the cavern with salt is also feasible with current technology, although it is not a normal practice in salt mining. However, even with back-filling, storage and disposal of mine tailings presents an unresolved serious ecological problem. A review of the problem has been commissioned by OWI (Reference 5-25).

Problems of mine design are somewhat different than those of conventional salt mines because of the mechanical and structural effects of the augmented thermal regime created by the presence of heat generating wastes (closure of excavated rooms due to increased plastic flow rates; thermal expansion of the salt mass). Theoretical models of the deformation behavior of bedded salt deposits are being constructed by both Sandia and OWI. These models can utilize laboratory data on mechanical behavior of both rock and mineral specimens in the temperature range of interest. They can also utilize field data gathered at Asse, in Project Salt Vault, and the planned Avery Island Heating experiment (Reference 5-25, page 134). The Avery Island in situ experiments consist of placing canned heaters in sleeves in dome salt to simulate the thermal characteristics of actual radioactive waste emplaced in a retrievable configuration.

The tests will provide specific information characterizing brine migration into the emplacement hole, temperature and temperature gradients in the adjacent salt, and the effects of elevated temperatures on salt during extended time periods. In addition, evaluations will be made of the corrosion between the salt and the carbon steel sleeves and the stability, during heating, of a drilled hole in salt with no backfill. In situ measurements of salt formation stresses, displacements, temperatures, and material properties will be obtained and can be used to verify the results of the rock mechanics analyses and to improve the modeling parameters used in such studies.

A further problem under investigation is that of keeping the shaft dry. To connect the repository to the surface it is necessary to sink several shafts, which will of necessity penetrate aquifers above the repository. The shaft must be sealed against leakage, so that no hydraulic connection can be established between the aquifer and the salt beds. Shaft leakage is presumably the cause of the flooding of abandoned salt mines near Asse. The problem is being studied by OWI, by Sandia, and at Asse. (References 5-25, 5-26, 5-30, 5-31). A related problem is borehole plugging. Boreholes have already been or will be drilled at all prospective repository sites, and must be plugged to prevent access of water to the salt deposit. This problem is under intensive study by OWI, and is not considered solved.

5.3.4.1.3 Experience in Storage of Radioactive Material in Salt Deposits.

Radioactive material has been stored in salt deposits in two R&D projects --one in West Germany and one in the United States. Both of these projects have utilized existing abandoned salt mines. Their performance provides an initial data base useful in evaluating transport, handling, emplacement and recovery of radioactive materials, and in evaluating the short term integrity of repositories. Furthermore, data collected on chemical effects, mechanical behavior and the thermal regime in the host rock during these operations provides a valuable check on laboratory and theoretical studies, as well as a source of data for numerical models. Some features of the R&D project are given below, further details are available in selected reports (Reference 5-25).

Project Salt Vault was an experimental study carried out by AEC/ORNL in an abandoned salt mine at Lyons, Kansas. The objectives of the experiment were to:

- (1) Test the feasibility and safety of nuclear waste handling equipment and techniques.
- (2) Determine the stability of salt under the influence of heat and radiation.
- (3) Collect information on the creep and flow of salt.

Over the course of 18 months, 21 canisters (each containing two irradiated fuel assemblies) were transported from the National Test Reactor Site in Idaho to the mine site in Lyons, Kansas, installed in holes in the floor of the salt mine, in some cases transferred from hole to hole, and finally removed from the mine. Concurrently, radioactive waste simulation tests were being conducted with electrical heaters. Valuable information on effects on the long-term stability caused by the presence of radioactive and heat generating wastes was accumulated (References 5-32, 5-33).

In West Germany, the Asse salt mine has served as a repository for low level wastes since 1967, as a repository for intermediate level since 1972, and as a test facility for the storage of high level wastes since late 1976. In addition, a conventionally solution mined prototype cavern is under construction at Asse, and is slated to receive intermediate level wastes in 1979 (Reference 5-30). While detailed technical information

is not generally available, it is clear that data collected during the continued operation of this mine on a repository will be valuable in evaluating all aspects of the long-term stability problem in salt caverns.

The Asse Salt Mine, however, is not in a bedded salt deposit. The original horizontal sedimentary layering that defines bedded salt deposits has been distorted by upwelling of the salt to form a salt anticline. While many of the problems in evaluating long term site stability are the same for flat lying salt beds, salt anticlines and salt domes, there are differences in stratigraphy, hydrologic regime and tectonic setting.

5.3.4.1.4 Rock-Radioactive Waste Interactions. Storage of radioactive materials in bedded salt has a number of effects which have a bearing on the long term integrity of the waste-salt system. These effects may be categorized as thermal, mechanical, and chemical, but these categories are not mutually independent.

The most obvious of these effects is an increase in host rock temperature caused by the presence of radioactive wastes. Any analysis of repository design and ultimate waste storage capacity will require the prediction of the temperature distribution as a function of time. In addition, many processes and properties are thermally driven or influenced and cannot be analyzed thoroughly without detailed knowledge of the thermal regime. The early development work in thermal analysis and experimentation has been conducted, but more work is required to develop thermal design criteria for the waste. These criteria, however, cannot be developed without consideration of the mechanical, chemical, experimental and overall system requirement influences. An iterative process is envisioned before adequate and acceptable criteria can be established.

Two field studies, Project Salt Vault and Asse, have yielded numerical data which have been reproduced by modeling. A third study, the Avery Island Heating Experiment, is planned by OWI (Reference 5-25, p. 34). Numerous model studies have been made. Despite the inherent difficulty in making the calculations, and the need for data acquisition of the thermal properties of the rocks present at each site, the problem of determining the thermal characteristics of bedded salt is tractable.

Thermally driven processes that might lead to the introduction of water or brine into the disposal site include:

- (1) brine migration,
- (2) Decrepitation of brine inclusions,
- (3) thermal decomposition of hydrous salts,
- (4) dehydration of shale partings (shale interbedded with the salt),

Brine migration, the migration of natural inclusions of brine within salt crystals up a thermal gradient, has been analyzed in single crystals in

the laboratory (Reference 5-34). Data from Project Salt Vault have been modeled fairly successfully. Field studies on the volumetric importance of brine migration are presumably being carried out at Asse, and will be included as part of the experimental work carried out during the Avery Island Heating Experiment. The process may be reasonably well understood; however, its quantitative importance in geologic as opposed to laboratory materials is not as well understood. If the repository is operated in a retrievable mode for 10 years or more, brine inclusions that migrate to the canister walls can be removed as vapor by the ventilation systems.

Decrepitation of brine inclusions is the vaporization and explosion of natural brine inclusion upon heating. Calculations of any effects of this upon either quantities of brine released and material strength properties have not been made. Decrepitation does not occur until a certain threshold temperature, (a function of brine inclusion composition, size, and host crystal strength) is reached. A threshold temperature of approximately 250°C at one atmosphere for brine in halite is generally cited, so decrepitation may be avoided by maintenance of temperatures below this critical value. (See Section 6.4.)

Several salts often present in evaporite assemblages contain structural water, and are unstable at low temperatures. At Asse, a thick seam of carnallite ($\text{KCl} \cdot \text{MgCl}_2 \cdot 6\text{H}_2\text{O}$), which decomposes at 110°C with attendant release of water, is present in the evaporite section. Any analysis of the potential effects of the dehydration of hydrated salts is very much site dependent, since mineralogy, quantity and the three dimensional distribution of such minerals are not predictable a priori.

The importance of the dehydration of shale will vary from site to site. Project Salt Vault showed that operation of electrical heating elements simulating radioactive waste in the vicinity of shale partings caused inflow of water to the heating elements.

The long term effect of the introduction of modest quantities of water or brine into the immediate vicinity of the waste containers is not clear. Radiolysis of the brine could occur with the production of gases and chemical species in solution (hydrochloric acid, chlorates and perchlorates, hydrogen). The nature of these reactions and of these products identify physical uncertainties and the need for research to answer questions which may influence repository design and environmental control. Additional concern is stated regarding the design and lifetime of the canister. Short term effects may include some dispersion of radioactive wastes, enhanced corrosion of containers, and problems in operation, maintenance, and retrievability.

The amount of water produced is a function of amount of water available and the thermal history, and will decrease with time as the above-listed sources are tapped. Water produced in situ may pose no long term effect, having been removed during routine repository operation.

Storage of energy in halite crystals as a result of gamma ray exposure and radiolysis of halite resulting in the production of chlorine gas have both been proposed as potential problems, but laboratory and field studies thus far show them to be of little consequence (Reference 5-25).

Mechanical effects are related to material properties as a function of temperature, as was discussed briefly before.

5.3.4.1.5 Release and Transport of Radionuclides to the Biosphere.

Since the main barrier to the release of emplaced radionuclides into the biosphere is to be long-term isolation in bedded salt deposits, it is necessary to examine all ways in which this isolation could be breached. Assuming loss of isolation, the pathways from the waste to the biosphere must be evaluated in terms of rates of release and transport of the various radionuclides, and modes of incorporation into the biosphere.

A number of scenarios can be constructed in which connection between the repository and the biosphere is established:

- (1) Natural dissolution of the salt beds by the action of circulating ground water. This process must be looked at in connection with all other scenarios.
- (2) Establishment of hydraulic connections between the overlying aquifer(s) and the disposal horizon by failure of mine shaft or borehold sealing, or faulting and/or displacement upward of the disposal horizon.
- (3) Establishment of hydraulic connection between overlying and underlying aquifers resulting from failure of borehole plugging or faulting.

Problems in evaluation of probability, time scale, and importance of these events stem from a geological and hydrological data base that is at present inadequate to model the present and future groundwater regimes, and rates of salt dissolution. Sandia has produced a reasonably detailed assessment of the probabilities and consequences of such events for the WIPP site (Reference 5-28).

Transport of radionuclides in groundwater is being examined in a number of ways:

- (1) Laboratory studies on the interactions of radionuclides with both individual minerals and rocks (pulverized and drill core samples)
- (2) Construction of theoretical models
- (3) Examination of behavior of radionuclides in the field, at the Hanford site, the Nevada Test Site, and the Oklo, Gabon natural reactor site.

This problem is complicated and is still in the analysis and data acquisition stage; models that have been constructed are one dimensional and simplistic. (See Section 6.4.)

REFERENCES

- 5-1 Legler, B. M. and Bray, G. R., "Concentration and Storage of High-level Liquid Waste at Allied General Nuclear Services", presented at 68th Annual Meeting, A. I. Ch. E., Los Angeles, California. Nov. 16-20, 1975.
- 5-2 Fleischman, R. M. and Liikala, R. C. "Isotopic Composition of Radiologic Properties of Uranium in Selected Fuel Cycles", BNWL-SA-5375, Battelle Pacific Northwest Laboratories, April 1975.
- 5-3 Radioactive Waste Management Practices in Western Europe, European Nuclear Energy Agency, Organization for Economic Cooperation and Development, September 1971.
- 5-4 Lenneman, W. L., "Management of Radioactive Aqueous Wastes from U. S. Atomic Energy Commission's Fuel Reprocessing Operations, Experience, and Planning", Symposium on Management of Radioactive Wastes from Fuel Reprocessing, OECD and IAEA, Paris, 27 November-1 December 1972.
- 5-5 Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle, NUREG-0116, (October 1976).
- 5-6 Lloyd, M. H., "Instabilities and Solids Formation in Lwr Reprocessing Solutions", Trans. A.N.S., 24, 233, 1976.
- 5-7 "Policy Related to the Siting of Commercial Fuel Reprocessing Plants and Related Waste Management Facilities", Code of Federal Regulations, 10 CFR 50, Appendix F, U. S. Government Printing Office, Washington, D. C., 1972.
- 5-8 McElroy, J. L. Bonner, W. F., and Mendel, J. E. "Waste Solidification U.S.A.", BNWL-SA-5873, paper for presentation at the International Symposium on the Management of Wastes from the LWR Fuel Cycle, Denver, Colorado, July 12-16, 1976.
- 5-9 Mendel, J. E., "High-level Waste Glass", Battelle-Northwest, paper presented at the 22nd Annual Meeting of the American Nuclear Society, Toronto, Canada, June 16, 1976.
- 5-10 Mendel, J. E. and McElroy, J. L., Waste Solidification Program, Volume 10, Evaluation of Solidified Waste Products, BNWL-1666, Battelle Pacific Northwest Laboratories, Richland, WA 99352, July 1972.
- 5-11 Alternatives for Managing Wastes from Reactors and Post-fission Operations in the LWR Fuel Cycle. ERDA-76-43, United States Energy Research and Development Administration, Report Coordinated by Battelle Pacific Northwest Laboratories, Vol. 2, May 1976.
- 5-12 McElroy, J. L., Personal Correspondence, Chemical Technology Department, Battelle-Northwest, Richland, WA, May 1977.

- 5-13 Quarterly Progress Report, Research and Development Activities, Waste Fixation Program, April through June 1975, BNWL-1932, Battelle Pacific Northwest Laboratories, Richland, WA 99352, September 1975.
- 5-14 Long-term Plan for Management of ICPP High-level Wastes, ACI-204, Allied Chemical Corporation, Idaho Chemical Programs--Operations Office, Idaho National Engineering Laboratory, Idaho Falls, Idaho 83401, June 1976.
- 5-15 Siemans, D. H. and Bonner, W. F., "Design Features of a Full-scale High-level Waste Vitrification System", Proceedings of 24th Conference on Remote Systems Technology, Battelle Northwest Laboratories Paper, 1976.
- 5-16 Mendel, J. E. et al., Thermal and Radiation Effects on Borosilicate Waste Glasses, IAEA-SM-207/100, BNWL-SA-5534, Battelle-Northwest, paper for presentation at Vienna, Austria, March 22-26, 1976.
- 5-17 Mendel, J. E., "Ceramics in Nuclear Waste Management", BNWL-SA-5757, paper for presentation at the 12th State-of-the Art Symposium on Ceramics in the Service of Man, Washington, D. C. June 7-9, 1976.
- 5-18 Roberts, F. P., Jenks, G. H., and Bopp, C. D., Radiation Effects in Solidified High-level Wastes--Part 1, Stored Energy, BNWL-1944, Battelle Pacific Northwest Laboratories, Richland, WA 99352, January 1976.
- 5-19 Platt, A. M., Research and Development Activities, Waste Fixation Program, BNWL-1699, Quarterly Progress Report, July-November 1972, Battelle-Northwest, Richland, WA, December 1972.
- 5-20 Mendel, J. E., et al, Annual Report on the Characteristics of High-level Waste Glasses, BNWL-2252, Battelle Pacific Northwest Laboratories, Richland, WA 99352, June 1977.
- 5-21 Turcotte, R. P., Radiation Effects in Solidified High-level Wastes Part 2 - Helium Behavior, BNWL-2051, Battelle Pacific Northwest Laboratories, Richland, WA 99352, May 1976.
- 5-22 Ross, Wayne A., Development of Glass Formulations Containing High-level Nuclear Wastes, BNWL-SA-6071, Battelle Pacific Northwest Laboratories, Richland, WA 99352, May 1977.
- 5-23 Smith, T. H. and Ross, W. A., Impact Testing of Vitreous Simulated High-Level Waste in Cannisters, BNWL-1903, Battelle Pacific Northwest Laboratories, Richland, WA 99352, May 1975.
- 5-24 McElroy, J. L. and Ross, W. A., Personal Correspondence, Chemical Technology Department, Battelle-Northwest, Richland, WA 99352, July 1977.

- 5-25 National Waste Terminal Storage Program, Progress Report Y/OWI-8, prepared for Energy Research and Development Administration by Office of Waste Isolation, Union Carbide Corporation--Nuclear Division, Oak Ridge, Tennessee 37830, November 30, 1976.
- 5-26 Bishop, W. P. et al., Proposed Goals for Nuclear Waste Management, Report to the United States Nuclear Regulatory Commission, Draft (not dated).
- 5-27 Lieberman, J. A., Rodger, W. A., and Baranowski, F. P., High-level Waste Management, Testimony prepared for the Nuclear Industry and California Electric Utilities, for presentation before the Energy Resources Conservation and Development Commission, State of California, March 21, 1977.
- 5-28 Program Plan for the Development of the Waste Isolation Pilot Plant (WIPP), Staff, Waste Management Systems, Department 1140, Sandia Laboratories, Kirtland AFB, East Albuquerque, New Mexico 87115, Mar. 10, 1976.
- 5-29 Holdoway, K. A., "Behavior of Fluid Inclusions in Salt During Heating and Irradiation", Trans. Am. Nuclear Soc., Vol 15, No. 90-91, pp. 303-312, 1972.
- 5-30 Kuhn, K. and Hamstra, J., "Geologic Isolation of Radioactive Wastes in the Federal Republic of Germany and the Respective Program of the Netherlands", Proceedings of the International Symposium on the Management of Wastes from the LWR Fuel Cycle, sponsored by ERDA, Denver, Colorado, July 11-16, 1976. (available from NTIS, U. S. Dept. of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161)
- 5-31 Weart, W. D., "New Mexico Waste Isolation Pilot Plant, A Status Report", Proc. of the Symposium on Waste Management, Sponsored by Univ. of Arizona, Arizona Atomic Energy Commission, and Western Interstate Nuclear Board, CONF-761020, pp. 173-187, Tucson, Arizona, October 3-6, 1976.
- 5-32 McClain, W. C., Bradshaw, R. L., and Empson, F. M., Disposal of High-level Solidified Waste in Salt Mines, Health Physics Div., ORNL, Oak Ridge, Tennessee 37830.
- 5-33 Empson, F. M., Bradshaw, R. L., McClain, W. C., and Housa, B. L., "Results of the Operation of Project Salt Vault: A Demonstration of Disposal of High-level Radioactive Solids in Salt", Trans. Am. Nuclear Soc., pp. 455-462, 1972.
- 5-34 Anthony, T. R. and Cline, H. E., "Thermomigration of Liquid Droplets in Salt", Trans. Am. Nuclear Soc., Vol 15, No. 90-91, pp. 313-321, 1972.

- 5-35 English, T. D., An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High Level Waste Management, JPL Publication 77-59, Vols. I and II, Jet Propulsion Laboratory, Pasadena, California, August 12, 1977.
- 5-36 Federal Decisions in the Nuclear Fuel Cycles, draft prepared for the Federal Energy Resource Council by the Staff of the ERC Nuclear Subcommittee, Draft #2. (not dated).

SECTION 6

STATUS AND ISSUES RELEVANT TO COMPONENTS OF THE
SPENT FUEL STORAGE AND DISPOSAL SYSTEM

6.1 SPENT FUEL STORAGE AT REACTOR SITES

6.1.1 Refueling Preparations

The following description is based on the design and planned operation of the spent fuel storage facility at the San Onofre pressurized water reactors (PWR), units 2 and 3 (Reference 6-1). The San Onofre spent fuel handling and storage facility arrangement and operation are representative of PWRs in general. Specifically, the San Onofre Units are representative of the majority of reactors either operating or planned in California. (San Onofre 1, 2 and 3, Rancho Seco 1, Diablo Canyon 1 and 2, and the proposed Sundesert 1 and 2 are all PWRs. Humboldt Bay, built in 1963, was the only BWR facility in California.) The current BWR designs resemble the typical PWR spent fuel operations and facility arrangements (such as San Onofre) to a large extent. (As explained by Howard Friend and Keith Schwartztraber at the California Energy Commission Hearing of March 10, 1977. (Reference Docket No. 76-NL-1, pages 8 and 76.) The current BWR plant has the spent fuel storage basin located in an auxiliary building adjacent to the reactor building. The storage basin at a PWR is situated in a building devoted specifically to fuel storage also located adjacent to the reactor building. The major difference between PWR and previous BWR storage designs is that the BWR pools were located within the reactor containment building at an elevated level.

During the cooldown of the reactor, preparations are initiated for the refueling operation. In the case of a pressurized water reactor, PWR, (using San Onofre Units 2 and 3 as a reference base) (Reference 6-1 page 9.1-29 to 31) the first step is the removal of the missile shields. The missile shields, as well as all the major components of the refueling and spent fuel storage operation, are shown in Figure 6-1. The control element drive mechanisms (CEDMs) are disengaged and the reactor vessel head nuts and studs are then removed. The CEDM cooling shroud is disconnected and the vessel vent line are removed. Plugs are placed in the empty stud holes and the reactor vessel flange is sealed to the refueling pool floor in order to prevent water from entering the lower portion of the reactor vessel cavity. The refueling pool is filled with borated water as a backup criticality safety measure prior to the reactor head being removed to its storage position (Figure 6-1). The next major operation involves moving the upper grid structure from inside the reactor vessel to its storage location on the refueling pool floor. This is accomplished by disengaging the CEDM's from their control elements, withdrawing the in-core instrumentation, and then lifting the structure with a special lifting rig. Once these operations are completed (approximately 7 to 10 days) (Reference 6-2, page 2-260) the refueling of the core can begin.

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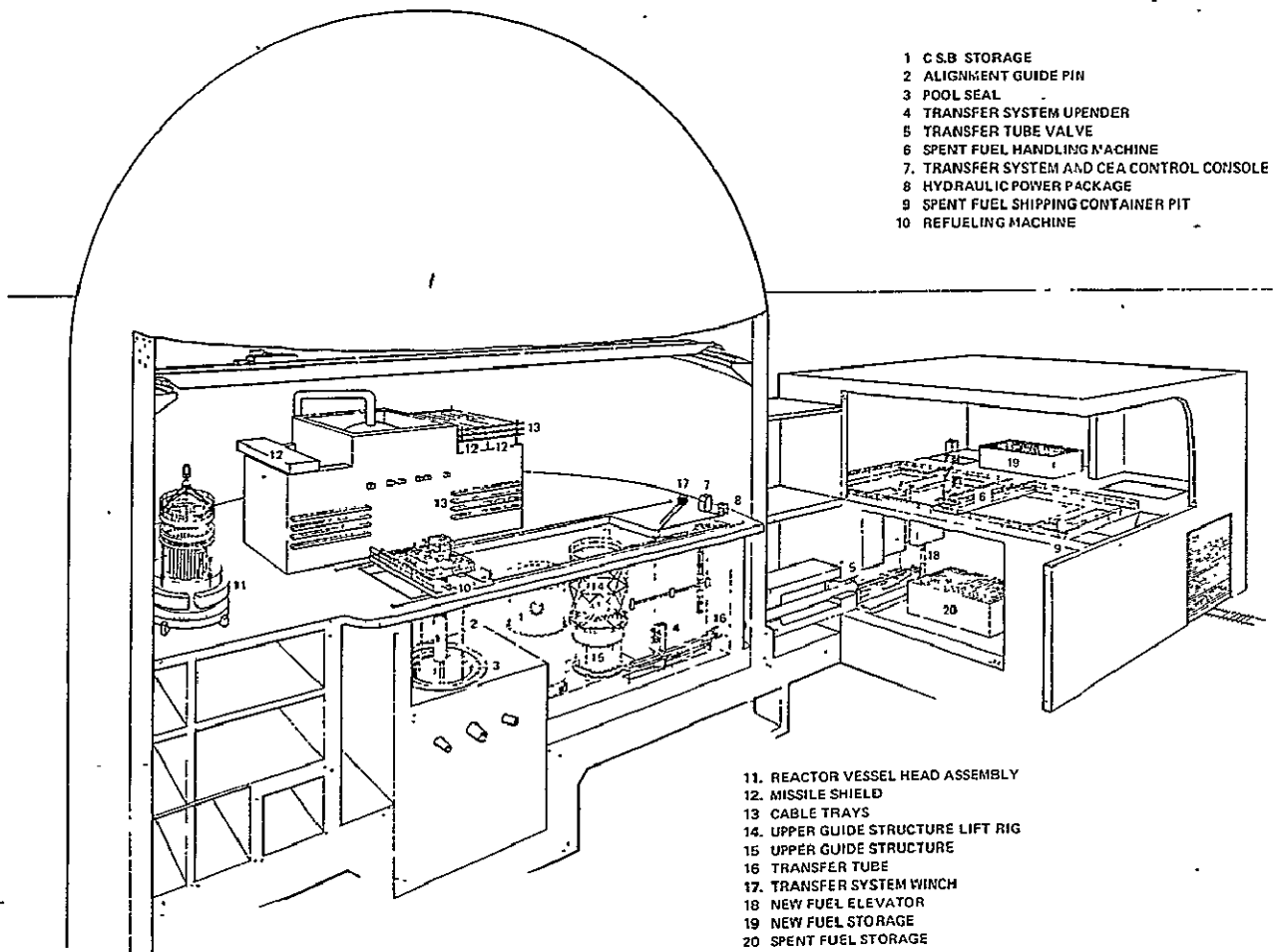


Figure 6-1. Arrangements of Major Components of the
Fuel Handling System (Ref. 6-1)

6.1.2 Removal and Storage of Spent Fuel

The refueling machine is utilized to remove spent fuel from the core and transport it to the transfer tube, to return new, or fresh, fuel from the transfer tube to the core, and to arrange fuel within the core to optimize fuel utilization. During movement, the fuel bundles are carried in the fuel hoist box to protect them from impacts and at a sufficient depth in the water to provide shielding for the refueling personnel.

Once the refueling machine reaches the transfer tube end of the refueling pool, the spent fuel bundle is discharged into a vertically standing transfer carriage (Figure 6-2). After the new fuel bundle is extracted from the second chamber of the transfer carriage, the carriage is then lowered to a horizontal position onto the transfer rails. The fact that the transfer carriage has two chambers permits both the refueling and spent fuel handling machines to travel loaded at all times--either to and from the spent fuel pool/new fuel elevator and transfer tube, or reactor and transfer tube.

The fuel bundle is transported through the tube to the spent fuel area (Figure 6-1) by a cable drive mechanism. At the spent fuel pool end the carriage is returned to the vertical position by the upending machine (Figure 6-2). The spent fuel handling machine transfers a new bundle from the new fuel elevator to the carriage and then moves the spent fuel bundle through the spent fuel pool/transfer canal gate to its designated rack position in the storage pool. The spent fuel handling machine not only transports the spent fuel from the transfer tube to the storage rack, but also moves the fuel from the racks and deposits it in a shipping cask located in the cask loading pit. Numerous interlocks and operational features are designed into the system to prevent damage to, or dropping of, spent fuel assemblies. On the handling machine hoist, interlocks are set to prevent attempting a lift if the load is too great and also to interrupt the operation if the cables go slack (Reference 6-1, page 9.1-25). Also travel stops on both the spent fuel handling and refueling machine prevent fuel assemblies from being hoisted to a point where less than 10 feet of water covers the active portion of the assembly. With 10 feet of water covering the fuel the radiation level at the water surface is limited to 2.5 mrem/hr (in the case of San Onofre 2+3) (Reference 6-1, page 9.1-34). The spent fuel handling machine has a translation interlock which provides speed restrictions on bridge and trolley movement unless the fuel is in the full-up position (Reference 6-1, page 6.1-20). Zone switches protect the handling machine from running the load into walls or the gate of the storage area (Reference 6-1, page 6.1-20). Manual operation of the spent fuel handling machine with direct visual contact by the operator of the fuel movement operation provides a backup factor in the event of an interlock failure. Redundant devices are provided in most cases to perform the same operation as the interlock and to provide the operator with information concerning the failure of the interlock (Reference 6-1, page 9.1-17). Also all major handling components are electrically interlocked with each other (Reference 6-1, page 9.1-33).

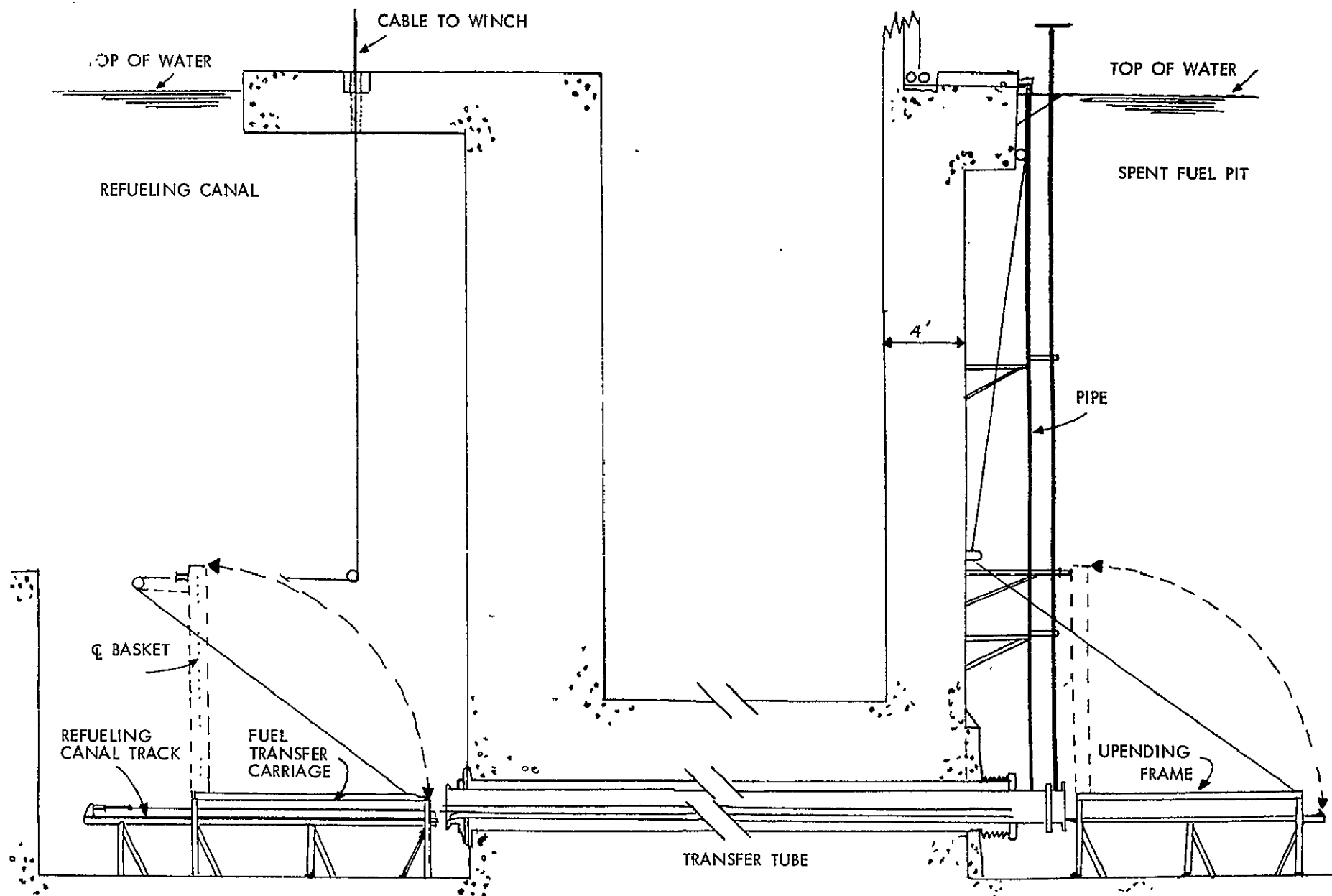


Figure 6-2. Fuel Transfer System (Ref. 6-3)

The hoisting of the fuel assemblies is performed once the grappling device is rotated by the actuator mechanism and engages the assembly. Inadvertent uncoupling of the assembly is prevented by positive locking of the grapple (Reference 6-1, page 9.1-25). During the withdrawing or insertion of the fuel assembly by the spent fuel handling machine, interlocks prevent any machine movement. Also during hoisting operations the load on the cable is monitored to ensure that the cable's movement is not restricted (Reference 6-1, page 9.1-25).

The controls for the spent fuel handling machine are located on the trolley of the handling machine. The location of the trolley is indicated by a pointer and target system while the coordinated location of the bridge is indicated by synchros on the console. In the event of a power loss the bridge, trolley, and winch can be operated manually by handwheels (Reference 6-1, page 9.1-25).

Prior to each fuel movement period recommended maintenance, including any necessary adjustments and calibrations, is performed (Reference 6-1, page 9.1-34). A dummy fuel assembly having the same weight, center of gravity, exterior size, and end geometry of an actual assembly is run through the equipment prior to each fuel loading operation as part of the preoperational testing sequence (Reference 6-1, page 9.1-33 and section 14.2).

Inservice inspections and tests, including checks of all control circuits, interlocks and alarm functions, are undertaken periodically on all components and systems (Reference 6-1, page 9.1-35).

During the reactor operation, the dry transfer tube is closed by a manually operated valve on the spent fuel pool side (transfer canal) and by a gasket flange inside the containment building (refueling pool) (Figure 6-2). For fuel transfer operations, the flange is removed and the refueling pool and transfer tube are filled with borated water. A common level in the spent fuel and refueling pools is reached prior to the valve being opened.

6.1.3 Loading and Preparation of Cask for Off-Site Shipping

Eventually, the assemblies will be shipped off the reactor site for storage or for reprocessing. The spent fuel shipping cask arrives at the loading area and is lifted off the railroad car or truck bed by a lifting beam. The overhead crane then moves the empty cask to the washdown laydown (decontamination) area.

Once serviced and washed, the cask is lowered into the spent fuel cask loading pit. The basic layout of the cask loading pit and spent fuel storage pool make it impossible to move the cask over the storage pool or storage racks -- the loading pit is the limit of the crane's movement (Reference 6-1, page 9.1-32). The first step of this procedure is the placing of the cask on an intermediate ledge of the pit to intercept the single lift over the fuel pool. The head

of the cask is removed at this point and the short rigging is replaced with long rigging for the final descent to the floor of the loading pit.

The spent fuel handling machine transports the fuel bundle underwater from the spent fuel pool rack through the pool gate and loads it into the shipping cask. The cask is lifted to the ledge where the head is reattached and the rigging again changed (this operation results in a maximum potential cask drop distance being 30 feet) (Reference 6-1, page 9.1-32). The cask is washed down as it is raised out of the pool and transported to the decontamination area where the radioactivity level on the exterior of the cask is reduced below the 10CFR71 limits (Reference 6-4). After the washdown is completed and the area is drained, the cask is moved to the shipping aisle and lowered onto the truck trailer or railroad car.

6.1.4 Development History for Reactor Pool Storage

Ever since nuclear power plants first began operation approximately 35 years ago, the storage of irradiated nuclear fuel in water basins (pools) has been standard practice. Originally, the water basin storage of irradiated (spent) fuel was considered the first of several short interim storage operations (i.e., at the reactor and at the reprocessing facility). But as a result of the current U.S. policy of indefinitely postponing reprocessing, reactor storage of spent fuel has currently become a potential long term interim storage option. Consequently, additional storage capacity will be required to meet this new situation.

Although some United States reactor fuel with zircalloy cladding has been in water basin storage for up to 12 years for some experimental stainless-clad fuel; up to 7 years for some commercial stainless-clad fuel; and approximately 20 years for some unirradiated stainless steel fixtures), water basin storage was not originally envisioned as a long-term operation (Reference 6-39, page 12-15). In the past, most nuclear power plants were designed on the assumption that the reprocessing sector of the fuel cycle would be in operation and that only a single batch (an annual discharge load) of spent fuel would need to be stored at the reactor site. This spent fuel was expected to remain at the reactor for less than a year (approximately 150 days) (Reference 6-3, I-96) before being shipped to a reprocessing facility. Therefore, a spent fuel pool storage capacity of 1 1/3 core was considered adequate for a single reactor--1 2/3 core capacity for two reactors sharing a pool. This would allow complete unloading of the reactor core, if necessary, for maintenance or inspection even if one batch (1/3 core load) was already in the pool.

In the early 1970s, the projections concerning the introduction of wide-scale spent fuel reprocessing began to change. For, with delays and cancellations in the operation of reprocessing facilities, it became evident that there would not be sufficient reprocessing capacity in the early 1970s. By late 1976, it appeared that reprocessing could not be foreseen before 1980-81. In April, 1977, President Carter postponed reprocessing indefinitely. The result is that an additional storage capacity for the irradiated (spent) fuel must be obtained. In 1974, this need for

increased pool capacity became apparent to the reactor operating utilities. By mid-1975, expansion (reracking) of the reactor storage pools had begun. This reracking operation involved the reduction of space between stored fuel bundles. As of June 1, 1977, 16 nuclear reactors had obtained NRC approval for this reracking operation. Another 18 reactors had submitted expansion requests to the NRC at that time (Reference 6-5) (This was out of a total of 63 operating nuclear power plants.)

6.1.5 Status and Plans

With the possibility of a shortage of spent fuel storage capacity because of the delays and cancellations of reprocessing operations, reactor operating utilities began to examine their available options around the end of 1973. Together, the NRC and the nuclear industry examined the three major options for extended spent fuel storage (Reference 6-5, page 4). One of these options is the construction and operation of independent spent fuel storage pools either at, or away from, a reactor site. Another option is the transferring of spent fuel from one reactor pool to another. The third option is the increasing of the storage capabilities of the existing pools.

The possibility of the construction and operation of independent storage pools either at, or away from, a reactor site is currently at a very preliminary level of review. One architect-engineering firm has discussed the development of an independent pool at a reactor site with the NRC. Another firm (Stone & Webster) (Reference 6-7) has produced a "topical report" on an independent spent fuel storage facility. In December 1974, the AEC issued a Regulatory Guide Number 3.24 (Reference 6-8) on independent spent fuel storage installations. Discussion within the literature and industry on the subject of independent storage pools (both at and away from reactor sites) has increased greatly within the last year. In any case, the development time frame for such independent pools is not expected to allow this option to assist in alleviating the urgent limited capacity situation within the next few years.

The transferring of fuel from one reactor storage pool to another has been approved only for interfacility movement on the same site (Reference 6-5, page 5). The NRC is presently reviewing the possibility of transferring spent fuel between reactor sites (Reference 6-5, page 5).

The third option, reracking, is the method utilities are expected to utilize first. To date, 30 applications (for 35 reactors) have been submitted to the NRC to increase the storage capacity of the original reactor pools. These requests vary from a 30% to a 250% increase in the original pool's capacity. The median request of 120% corresponds to an additional storage capability of 1 1/2 core's worth of fuel (Reference 6-5, pg. 6). This should provide another four years of annual discharge storage (1/3 of the fuel in the core is discharged per year). The cost is estimated to be between \$1-3 million for the entire operation (from design through installation of one pool rerack) (Reference 6-6, pg. 225). With respect to timing, current indications are that it will

require 1 1/2 to 3 years between design and installation (Reference 6-9, pg. 253). The present high-density rack design provides for a much tighter packing of the fuel assemblies than earlier high-density designs which reactor operators have either installed or are requesting NRC for permission to install. New rack designs allow for the possible storage of up to 8-12 years discharge (with a 1/3 core space reserve) depending on the type and configuration of the fuel and rack design (Reference 6-13, page 257, 6-12, page 258, 6-11, page 256). The result of this timing situation and the size of the expansion is that sometime around 1982-83 some of the reactor storage pools will again reach their capacity limits (while still retaining the fuel core discharge capabilities). Although future plants now being designed are being encouraged to consider enough capacity for the storage of a plant's lifetime quantity of irradiated fuel, there will still be a significant number of nuclear reactors in the mid-1980's that must look to one of the other options. The storage of spent fuel at an independent spent fuel pool and at a spent unprocessed fuel facility (SURFF) are examined in the following sections (see section 6.2 and 6.3).

6.1.6 Issues -- Engineering Achievability

The situation of storing irradiated nuclear fuel at reactor sites is currently in a period of transition. For, as mentioned, reactors have been operated with the belief that reprocessing services would be widely available and, therefore, a storage space of 1 1/3 core loadings would be adequate. But with reprocessing an uncertain prospect for the near future, all reactors are assumed to eventually request permission to expand their existing storage capacity. Because of the economics of the situation, the reactor operators are expected to rerack their existing pools as a first step in increasing their storage abilities. As a result, the major interest of those involved in spent fuel storage operations is presently directed at high density storage of irradiated fuel. Specifically, the NRC has shown particular interest in the criticality, cooling, and seismic analysis in addition to the design and quality assurance specifications and codes (Reference 6-10, page 586).

6.1.7 Design Approach, Constraints, Regulations

The general approach of those companies involved in designing reracking arrays or modules is either to establish a methodology which can be applied to a variety of fuel rack designs, or to produce a "standard" design independent of site conditions (References 6-11, 6-12, 6-13, 6-14). The objectives that must be met are to store fuel as close as possible while still maintaining a safe, subcritical array at a reasonable cost. But in designing a new rack system to meet these objectives and goals a number of constraining factors must be considered. The allowable storage capacity of a pool may be limited not only by the physical geometry of the pool, but also by the structural and seismic loads that can be imposed on the walls and floors. Also, as is the case of the BWR pools, a substantial increase in the storage capacity for irradiated fuel will require the replacement of some equipment in the pool for storage of other irradiated objects (such as control

rod blades, flow channels, and other BWR hardware). Finally, if a pool is already storing fuel, the new rack system must be installed without compromising the safety and in some other manner than being bolted to the floor -- such as being braced against the wall.

The racks are designed to meet ANSI N210 and use the ASME code Section III, Appendices 17 and F as a guideline for stress limits for normal and faulted conditions. The racks are classified as NRC seismic category I equipment and ANS safety class 3.

New high density fuel storage racks have been designed and constructed by each nuclear reactor vendor for their specific reactor fuel as well as by several independent companies for various customers (such as Exxon Nuclear--designed racks for Rancho Seco). The use of more accurate criticality codes, an allowable multiplication factor of 0.95 or less, and fixed poisons in the racks has reduced the centerline-to-centerline distance of the fuel in the racks for PWR assemblies from a previous value of about 21 inches to about 12 inches. For BWR high density racks, the reduction has been from about 10 inches to 8 inches (Reference 6-5, page 9).

Each rack is designed to hold from about 20 to 169 bundles depending on the fuel type (PWR or BWR), specific reactor design (General Electric, Babcock & Wilcox, Westinghouse, or Combustion Engineering) or fuel assembly outer envelope dimensions (Reference 6-11, page 256; 6-12, page 258; 6-13, page 257; 6-14, page 261).

6.2 INDEPENDENT SPENT FUEL POOL STORAGE

6.2.1 Process Description

The storage of irradiated nuclear fuel at an independent fuel storage facility consists of four operational phases.

- (1) The receipt of the shipping cask
- (2) The unloading of the cask
- (3) Movement into, and storage in, the storage basin of the spent fuel bundles
- (4) Preparation of the cask for reshipment
- (5) Shipment of spent fuel from Morris to a national repository will presumably occur in the future.

The following discussion of the above four phases is based on the operation of the General Electric Company's Morris Spent Fuel Storage Facility (the only currently operational central spent fuel storage facility in the United States (Reference 6-15)). The operation and basic design of the spent fuel storage operation at the GE Morris Operation is representative of the proposed Exxon storage system and the completed AGNS system. The design of the Morris Operation is also similar to that proposed by Stone & Webster for an independent spent fuel storage facility

(Reference 6-17). The details presented in the following sections refer to the GE Morris Operation but similar facilities, components, and operations will be provided for in any other independent spent fuel storage system. The cask used in this description is the IF-300 multi-element cask. Similar procedures are followed for smaller, single-element casks.

6.2.1.1 Cask Receipt. After entering the security fence gate, and before moving into the receiving area, the transport vehicle is stopped and the cask is inspected (Reference 6-15, page 1-17). A surface smear for radioactive contamination is performed and surface radiation levels are monitored. The vehicle then continues into the receiving area (see Figures 6-3, 6-4) and is prepared for transfer of the cask to the basin area. If decontamination of the cask is required, it is performed in a specifically designed decontamination location. If excess radioactivity is encountered on the cask surface, the appropriate reporting and radiation protection measures are taken. Minor damage to the cask might be repaired if the damage is within the limited maintenance capacity of the Morris Operation prior to unloading (Reference 6-15, page 5-12).

Once preparations are completed, the cask is lifted by a 125-ton capacity crane using an appropriate lifting yoke and is placed on the decontamination pad (Figure 6-5). The work stand is positioned around the upright cask and water and steam are used to remove road grime. The cask is flushed with demineralized water through shielded piping. This flush water is checked for radiation levels and is drained to the low activity waste vault (Reference 6-15, page 5-11). When the flushing is complete, an external radiation shield is attached at the parting line between the cask head and body to provide shielding from radiation streaming as the head is loosened. The nuts on the head are loosened (32 bolts on the IF-300 cask) (Reference 6-15, page 1-23) and all but four are removed. Cables are attached from the lifting yoke to the cask head for removing the head at a later time. The cask is then moved to the unloading pit.

The cask is lowered onto the unloading pit shelf and the remaining head bolts are removed (Reference 6-15, page 1-23). This shelf is protected from the impact of a dropped cask by two one-inch thick steel plates separated by energy absorbing fins as seen in the top of Figure 6-7. A long hook is attached and the cask is lowered to the floor of the unloading pit (about 48 feet below the water's surface) without getting the crane's cables or hook in the water (Reference 6-15, page 5-13 to 5-15). Once on the unloading pit floor, the lifting yoke is disengaged and the head is lifted by the four cables and moved to an out of the way storage position in the pit.

6.2.1.2 Cask Unloading. The unloading of the fuel is done using a 5-ton capacity bridge crane. A grapple is attached to a fuel bundle which is then hoisted out of the cask and lowered into the appropriate storage baskets (a PWR basket can hold 4 bundles, while a BWR basket can hold 9 bundles) (Reference 6-16, page 3-5). The basket is latched

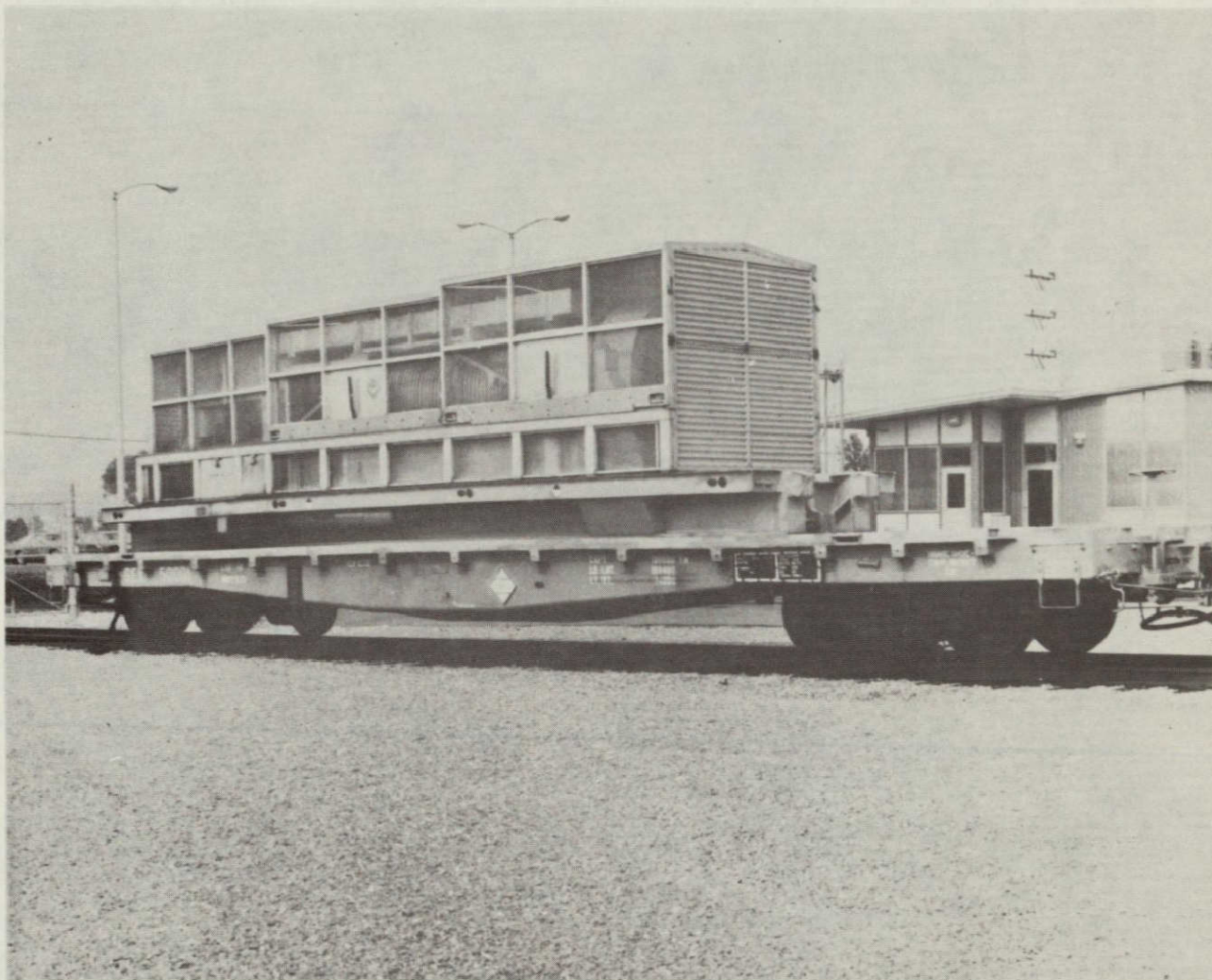


Figure 6-3. GE IF-300 Irradiated Fuel Shipping Cask: Mounted on a 100-ton railroad flat car, the IF-300 is in an enclosure assembly that provides redundant engine-blower air cooling, as well as screen barriers to prevent access to the cask surfaces. The IF-300 cask is designed to carry 7 PWR fuel assemblies, or 18 BWR fuel assemblies. The Morris Operation administration building is in the background. (Ref. 6-15) (Photo Courtesy of General Electric)

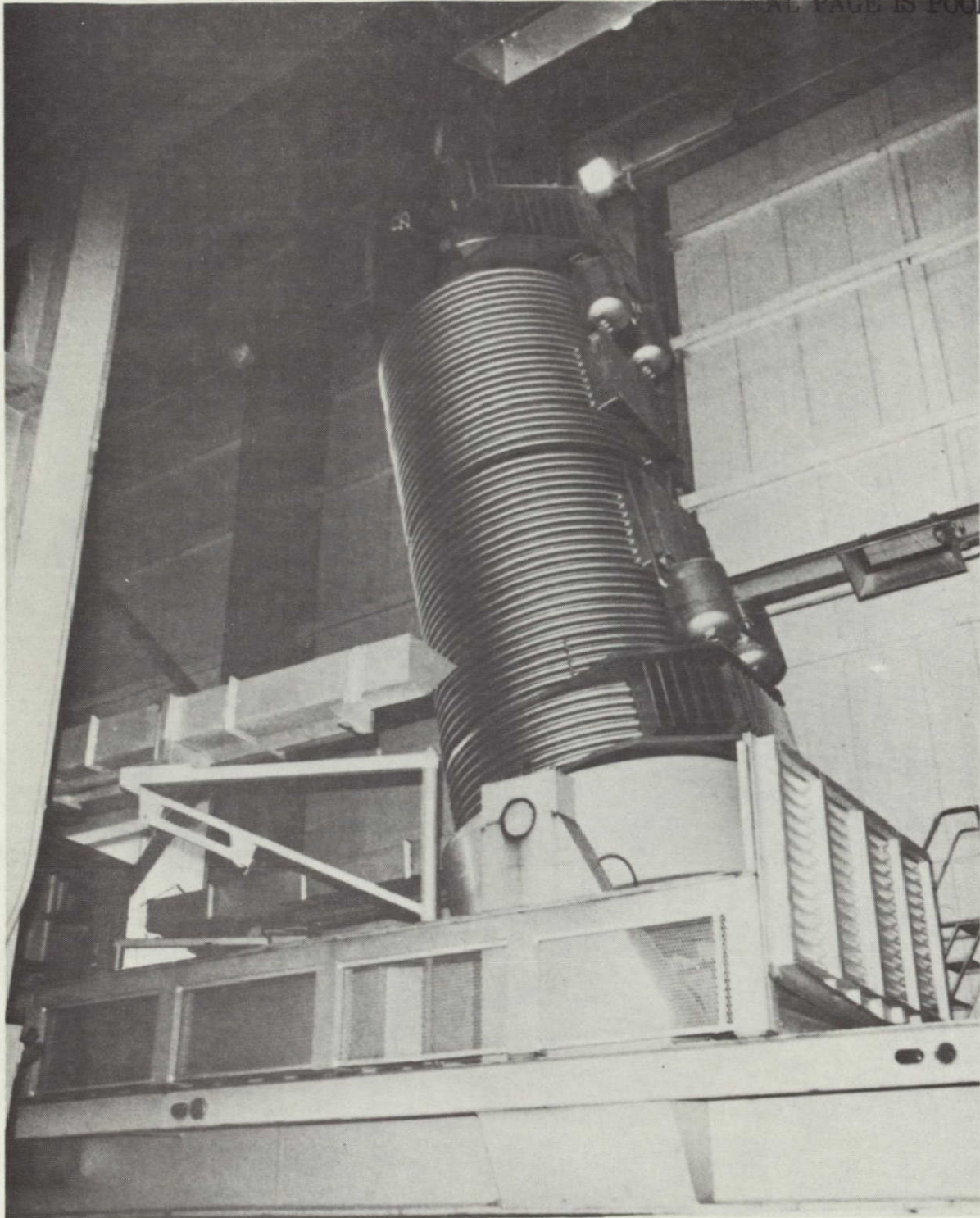


Figure 6-4. IF-300 Cask in Receiving Area: Preparations for lifting the cask from its transport vehicle are performed in the cask receiving area. In this photograph, the IF-300 cask is being lifted from the rail car skid by the 125-ton cask crane. The steel-frame, insulated metal siding and other details of the cask receiving area are visible. (Ref. 6-15) (Photo Courtesy of General Electric)

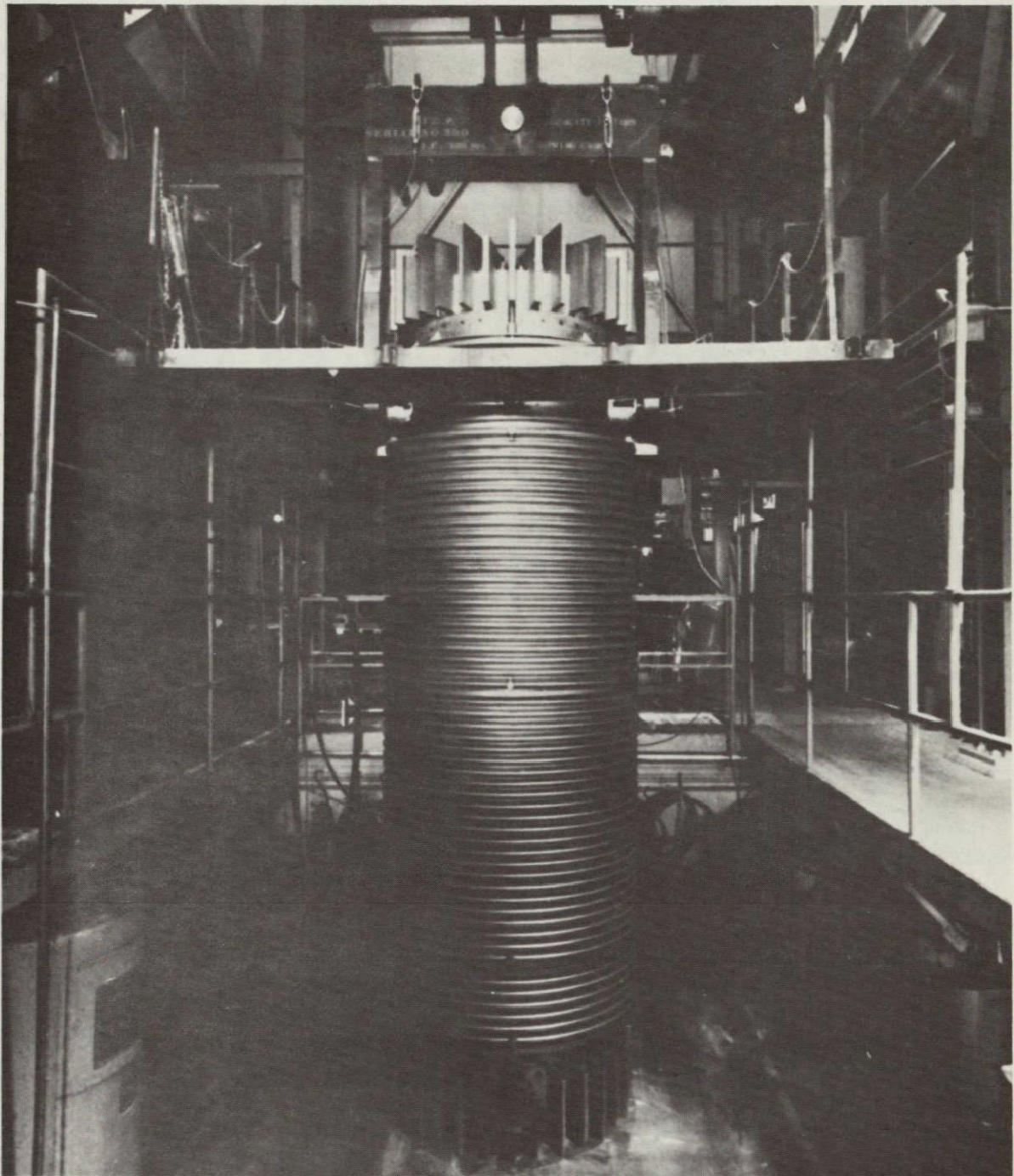


Figure 6-5. Cask Decontamination Area: The IF-300 cask is shown standing on the pad in the decontamination area with the lifting yoke attached (notice the slack head lifting cables). A work platform has been moved in place around the cask; different platforms are available to match the cask involved. Tankage contains detergents, part of the wash-down system. This view faces south, towards the cask unloading basin and fuel storage basins. The fuel-handling crane, mounted below the cask crane, can be seen in the upper right of the photographs. (Ref. 6-15) (Photo Courtesy of General Electric)

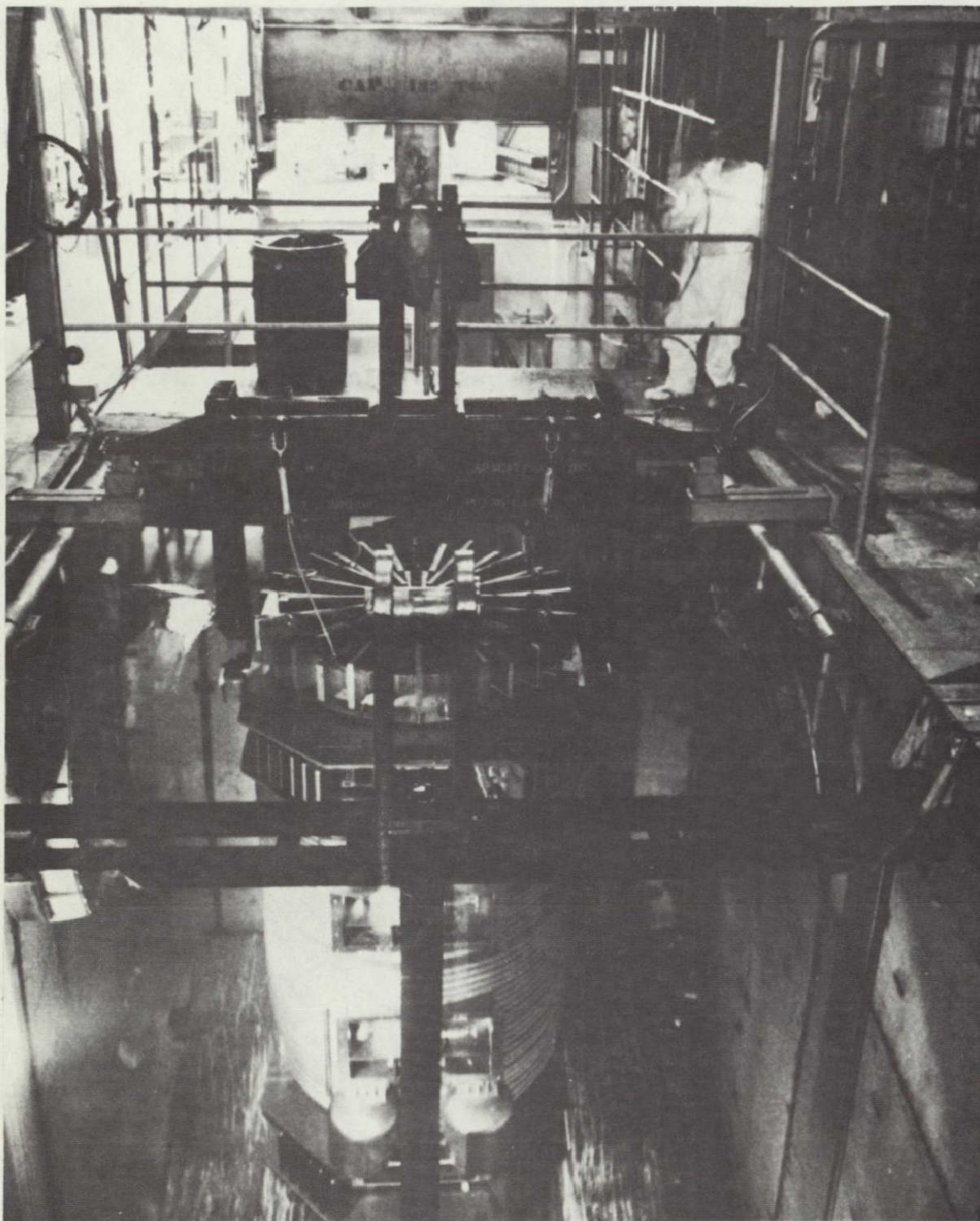


Figure 6-6. IF-300 Cask on Cask Shelf: Cask head closure nuts have been removed; the cask positioned on the shelf; the yoke disengaged from the cask trunnion; and the yoke placed on the fixtures extending from the basin walls. The extension hook (cross-shaped object in foreground) is also stored on wall fixtures. Note the crushable surface pad beneath the cask, used to absorb impact forces. (Ref. 6-15) (Photo Courtesy of General Electric)

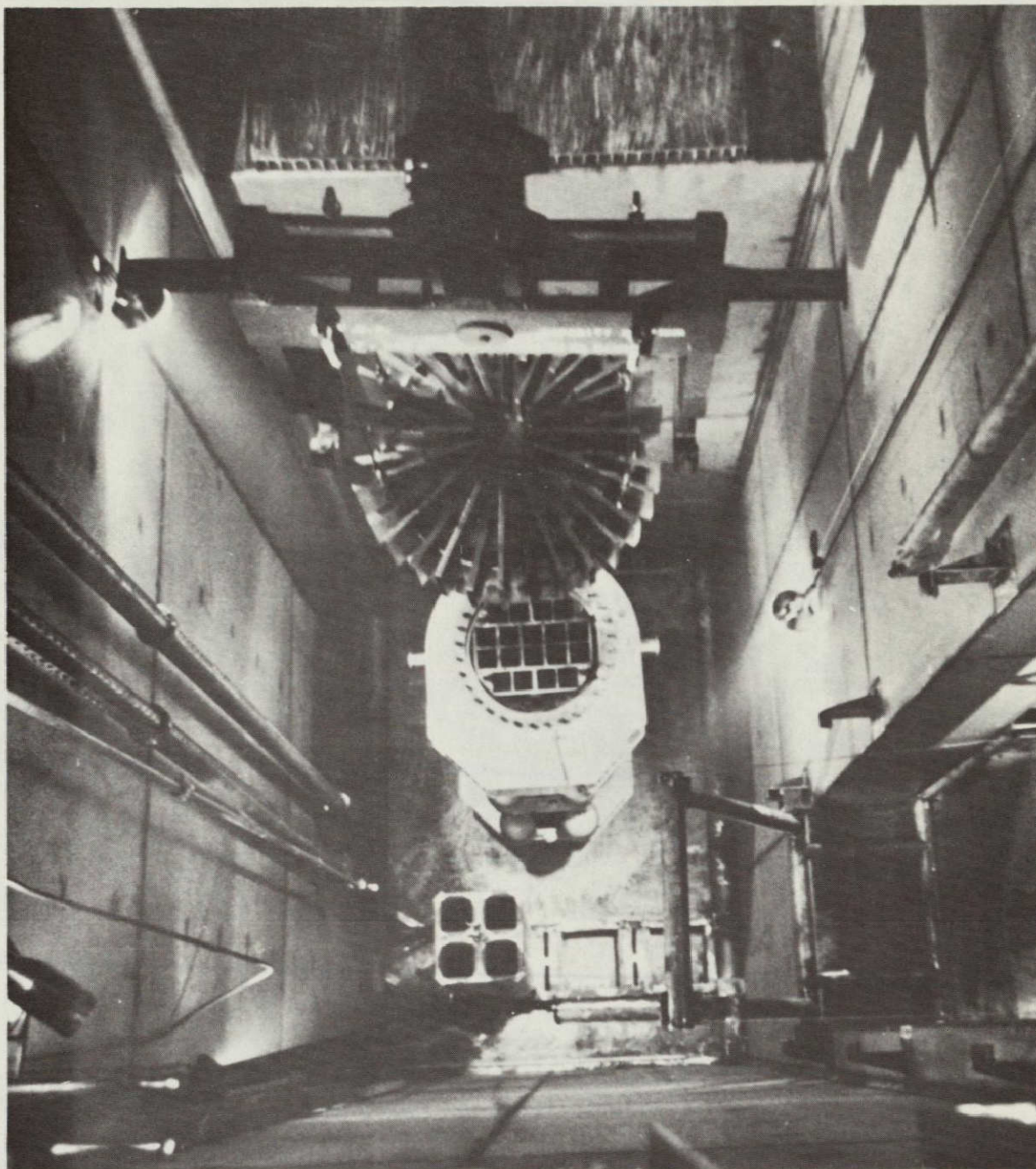


Figure 6-7. Cask Unloading Basin Deep Pit: The cask is in position on the floor of the deep pit as the yoke is being withdrawn, carrying the cask head supported by four cables. The lower right opening is the entrance to Basin 1; the open, cable-supported gate prevents fuel spills into the deep pit if a basket should fall in that direction (see Section 6.2.4). Three positions in the pit are available for fuel baskets; a single PWR basket is in position. The crushable, energy absorbing plate on the basin shelf is visible at the top. Baskets must be moved to the grid position below the gate prior to lifting from the deep pit. (Ref. 6-15) (Photo Courtesy of General Electric)

in the supporting grid nearest the transfer gate leading to the storage basin during loading (Figures 6-8 through 6-10). Once the loading of the basket is completed, it is transferred past the basin door guard (which prevents fuel from falling out of the basket in the event of a basket tip) and into the storage basin by another crane (Figure 6-11) (Reference 6-15, page 5-16 to 5-18).

6.2.1.3 Movement and Storage of the Fuel in the Basin. The upward force exerted by the basin crane on the basket releases the basket latch mechanism which fastens the basket to the supporting grid (Reference 6-15, page 5-20 to 5-27). The basket is lifted from the grid, carried through the gate and into the storage basin. The basket is lowered into its designated position and set in the stainless steel support grid. The upward force is released and the latching mechanism secures the basket to the grid (Figures 6-11 and 6-12).

6.2.1.4 Cask Reshipping Preparation. Once the cask has been emptied and the fuel transferred to its storage position, the cask is prepared for shipment back to the reactor facility. The cask head is lowered back on the body of the cask and the lifting yoke is reattached. The crane then lifts the cask to the unloading pit shelf and the extension hook is returned to its stowaway position. After the yoke is re-engaged to the shorter lifting crane hook, the cask is hoisted out of the water. The exterior of the cask is rinsed with demineralized water as the cask clears the water (Reference 6-15, page 1-26). The water in the interior of the cask is drained into the basin. After further rinsing, the cask is moved to the decontamination pad. The head nuts are inserted and tightened, a smear test for radiation contamination is made for a radiation check, and the cask is loaded back onto the transport vehicle (Reference 6-15, page 1-26).

6.2.1.5 Spent Fuel Reshipment. The eventual reshipment of the spent fuel from the Morris Operation to a national repository will be accomplished utilizing much the same procedures and checks as were required in the receipt, unloading, and movement of the fuel. One of the differences in reshipping an empty cask and one loaded with spent fuel is that the cooling water inside a loaded cask must be sampled for excess radiation -- a step requiring a 6 to 10 hour delay after loading (Reference 6-41, page 10). Other fuel reshipment operations would be the same (but in reverse order) as those previously described for receipt of the fuel.

6.2.2 Development History

Although other storage pools located away from reactors have been constructed for the holding of spent fuel, only the General Electric Company's Morris Operation has been operated as a spent fuel storage facility independent of any other fuel cycle process. The two other non-reactor pools constructed for commercial spent fuel storage are at the AGNS and NFS reprocessing facilities. The NFS pool has been used in the

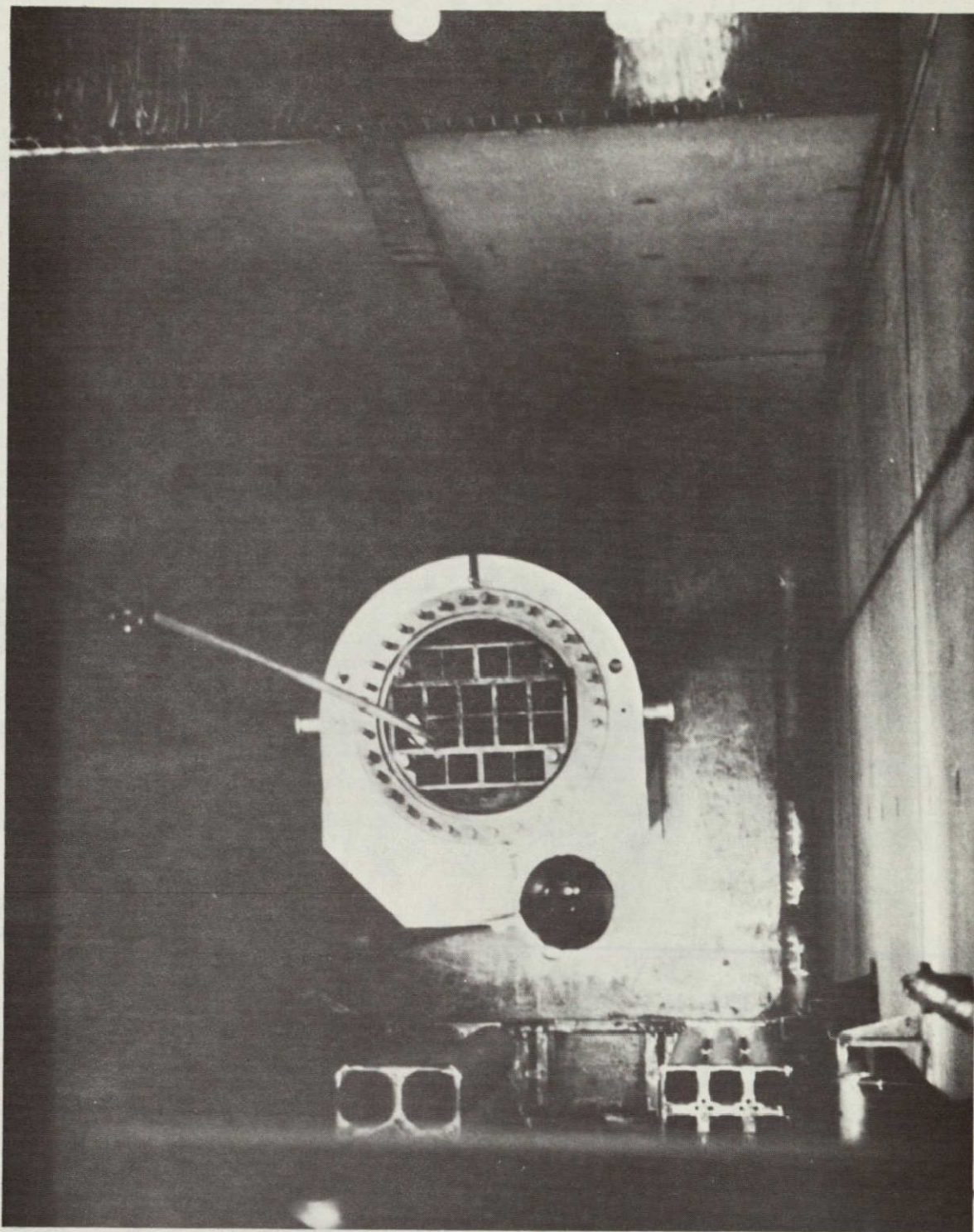


Figure 6-8. Fuel Grappling Operations: Viewed from south wall of the deep pit, a grapple has been attached to the fuel handling crane and has engaged the bail of a BWR fuel bundle in the cask. At the lower right, a partially filled BWR fuel basket is in position. Underwater lights (foreground) and an underwater closed-circuit TV are available to facilitate fuel transfer operations. (Ref. 6-15) (Photo Courtesy of General Electric)

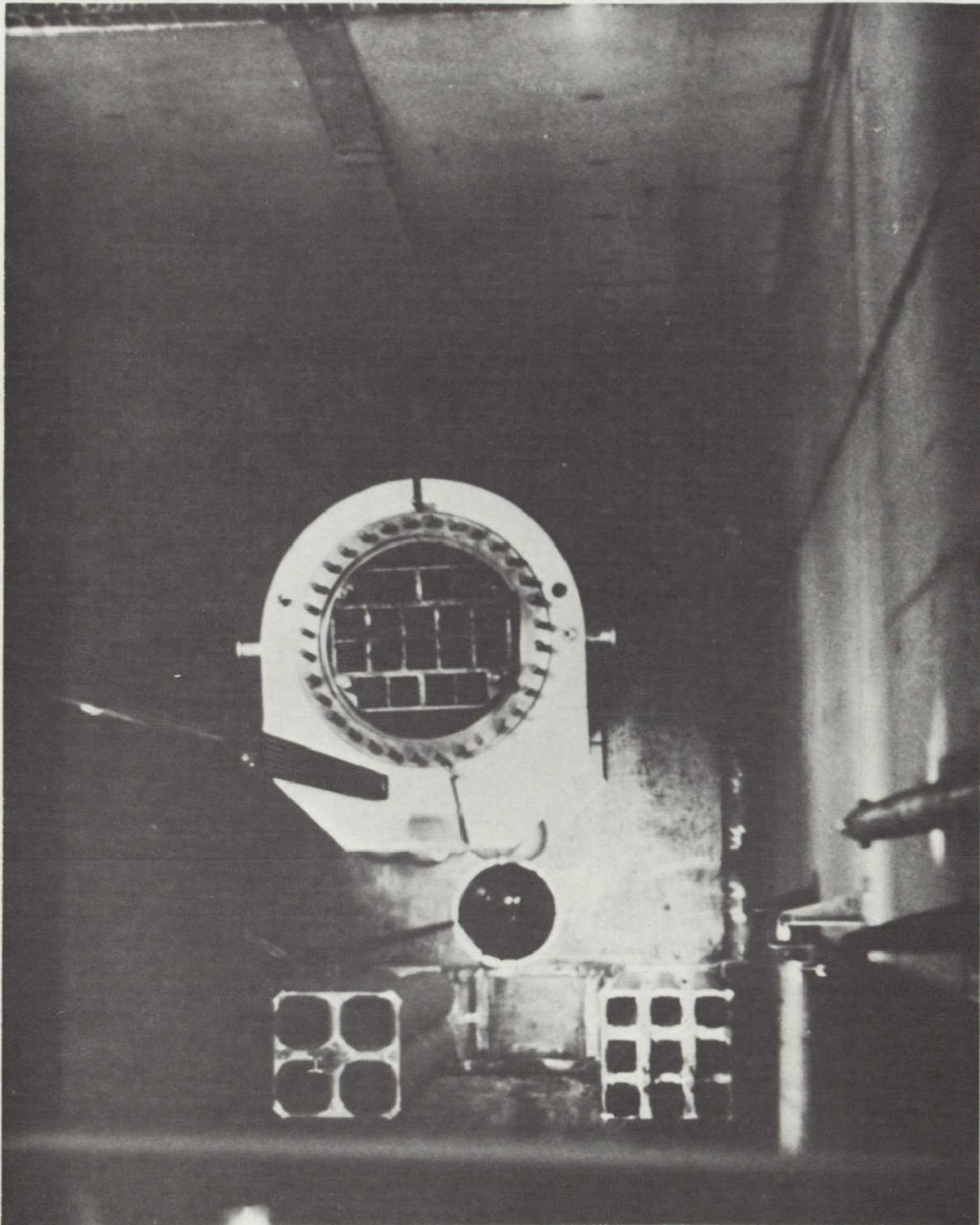


Figure 6-9. Fuel Removal from Cask: The grappled fuel bundle is lifted from the cask and carried towards the fuel basket at lower right. Notice that the basin doorway guard is in the upright position; this is necessary during transfer operations. The length of the grapple and hoist height limit prevents lifting the fuel bundle above the depth of water required for shielding. (Ref. 6-15) (Photo Courtesy of General Electric)

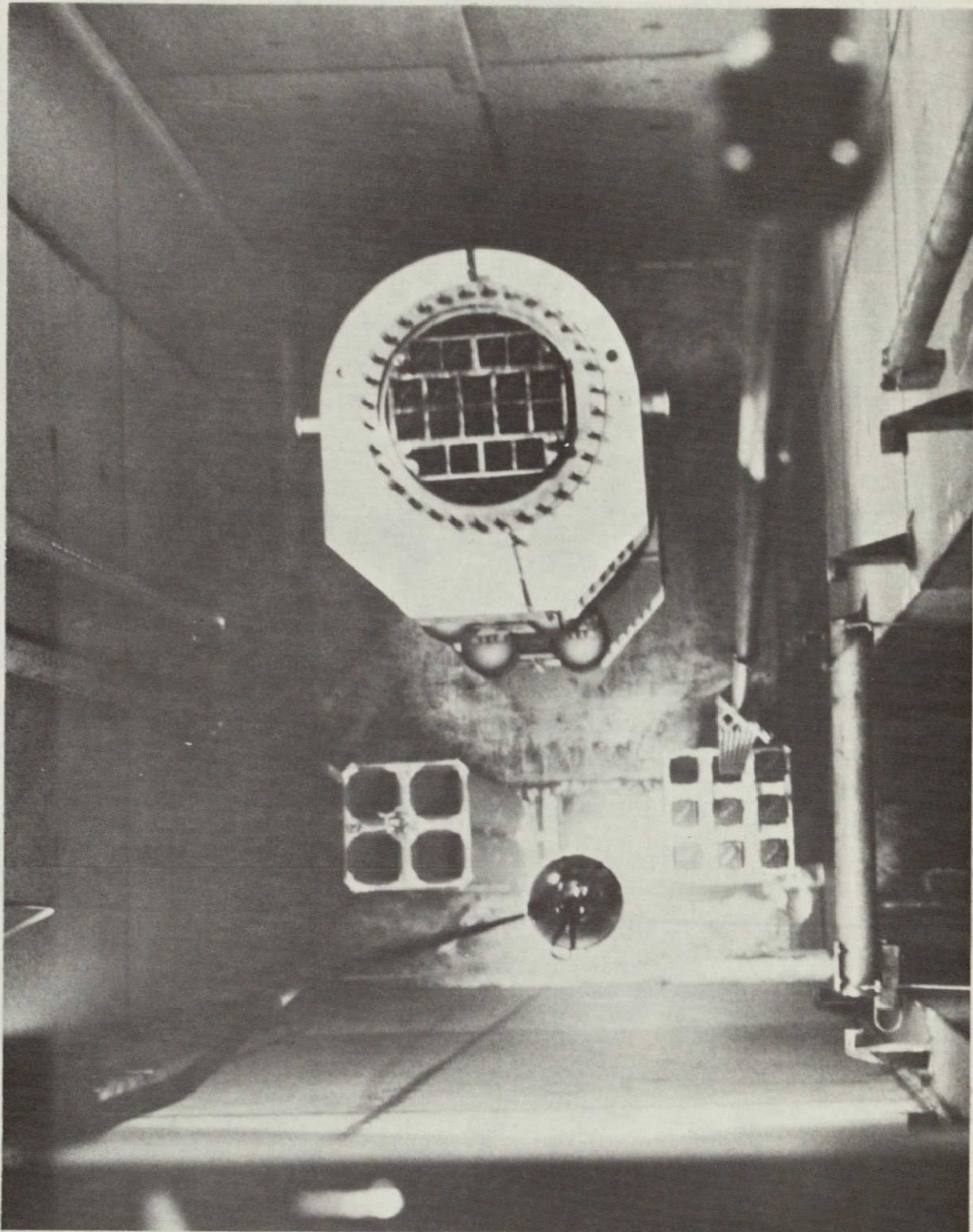


Figure 6-10. Fuel Basket Loading: The bundle is positioned over the basket, and lowered. Although baskets are designed to hold the bundles without significant lateral movement within the basket, the basket openings are large enough for easy insertion of the bundle. These operations continue until the basket(s) are full, or the cask is empty. (Ref. 6-15) (Photo Courtesy of General Electric)

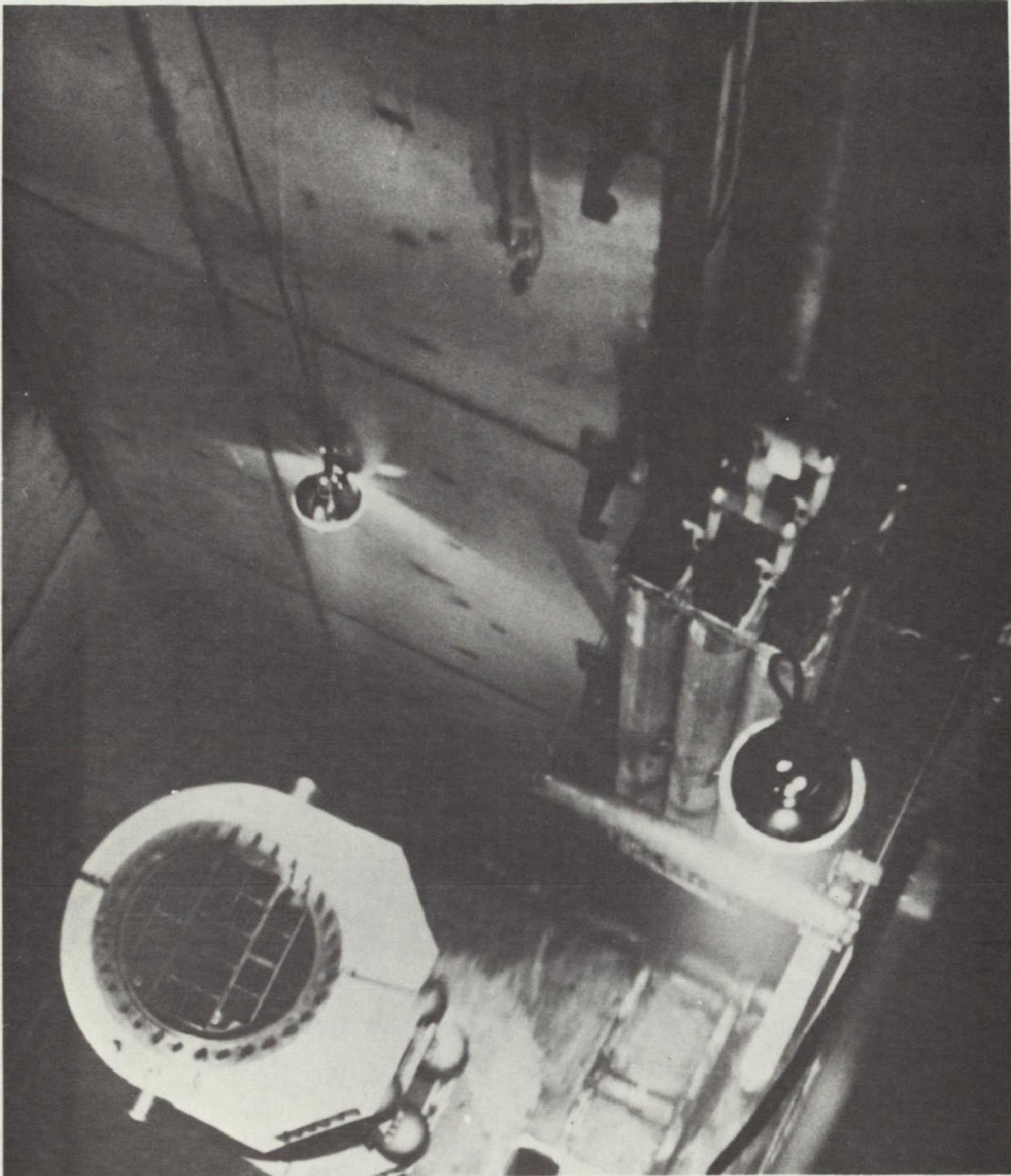


Figure 6-11. Fuel Basket Transfer: To transfer a loaded basket from the cask unloading pit the cask must be in the east support grid position, and the fuel basin door guard must be in the position shown (lower or "down" position). The basket grapple, attached to the basin crane, engages the basket lifting rods. Upward pressure by the crane releases the basket latch mechanism, unlocking the basket from the support grid. The basket is lifted upward, through the door guard opening, and moved into the storage basin area. (Ref. 6-15) (Photo Courtesy of General Electric)

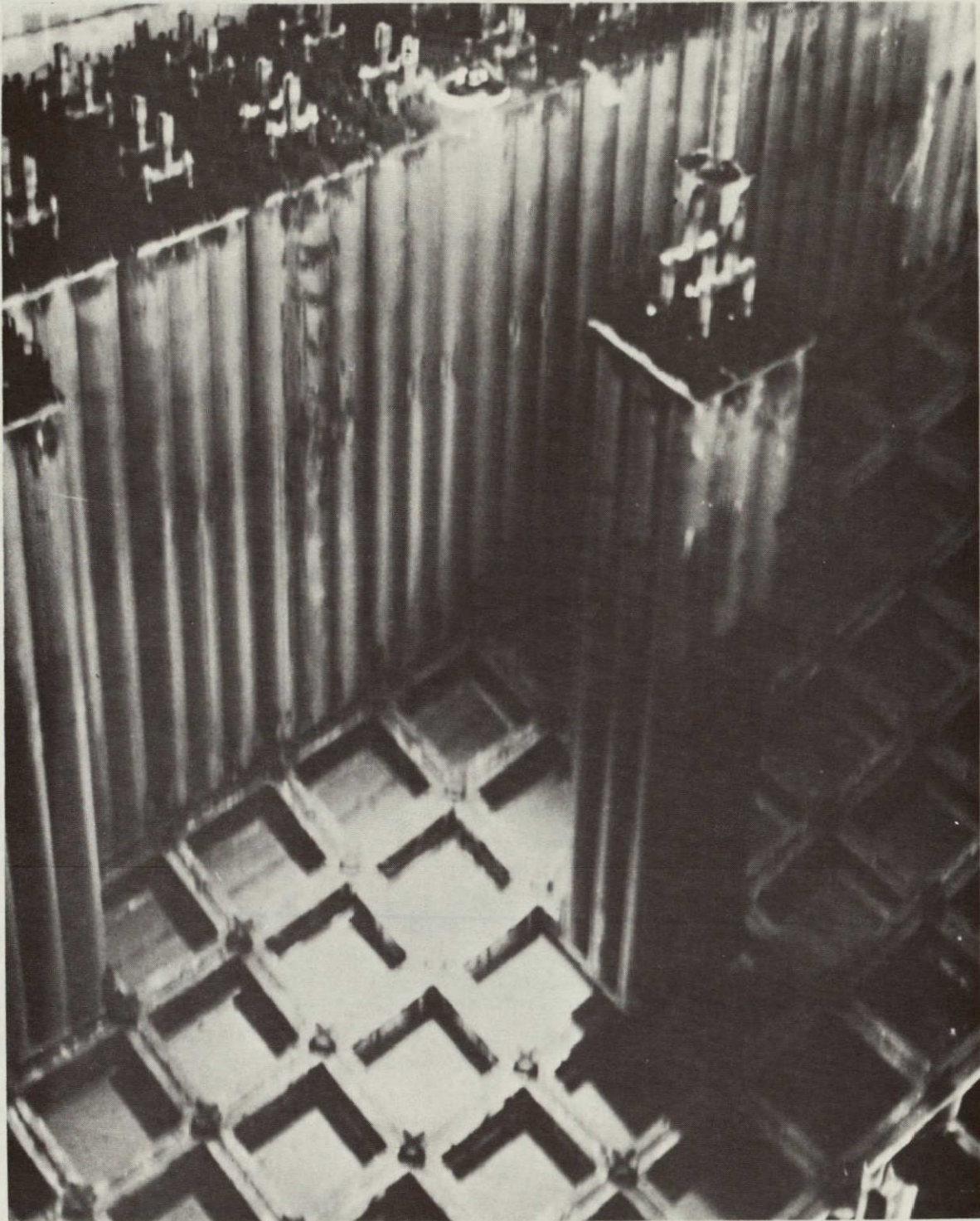


Figure 6-12. Fuel Movement In Storage Basins: The fuel basket is carried by the basin crane to a designated position in either Basin 1 or 2. The basket is positioned in the stainless steel support grid and when upward force on the basket lifting rods is released, the basket is locked in the grid. Fuel storage patterns in the basins are determined by administrative procedures. (Ref. 6-15) (Photo Courtesy of General Electric)

past for interim storage of spent fuel prior to reprocessing. The AGNS pool has not been issued a license for receipt and storage of irradiated commercial nuclear fuel. As a consequence, the GE Morris Operation is used in this report as the basis for (current and future) design, operation, and safety information concerning independent spent fuel storage pools--either located at a reactor site or at a central location for multi-reactor usage.

The Morris Operation was built as an integral part of the General Electric Midwest Fuel Recovery Plant (MFRP) and was licensed for receipt of spent fuel in December 1971. Although the Atomic Energy Commission subsequently terminated the MFRP Construction Permit (No. CPCS-3), in August 1974 the Materials License (SNM-1265) for the receipt and storage of spent fuel at the Morris facility was continued. In 1975 the license was amended to increase the storage capacity from 100 MTU to 700-750 MTU (dependent on the ratio of BWR and PWR fuel in storage) (Reference 6-15, page A.1-1 to A.1-3) by utilizing the former high-level waste canister storage basin for fuel storage. The Morris facility storage capabilities are provided under certain conditions to former GE reprocessing customers who face a threat of shutdown due to the lack of storage space because of the inoperability of the Morris reprocessing plant. The storage capacity is currently not available for general utility usage (Reference 6-40, page 6). It appears with the reracking being undertaken or planned that the existing facilities at Morris will serve GE's need and those of their original customers until about 1980.

6.2.3 Status and Plans

There are presently two tentatively planned away-from-reactor storage facilities. The first is the expansion of the GE Morris facility by 1100 MTU capacity through the addition of a third basin. The second project is the spent fuel pool associated with the proposed Exxon reprocessing facility. (To be constructed only in connection with a reprocessing facility.) (Reference 6-40, page 8.) The Exxon capacity would initially provide space for 3500 MTU with a possibility of expanding it to 7000 MTU thereafter, if needed. General Electric has stated that they are currently holding discussions with a few original reprocessing customers to explore possible arrangements for a joint venture to construct their expanded facility (Reference 6-42, page 2). GE indicates that they have no current plans to offer commercial fuel storage (Reference 6-40, page 6).

The Morris Operation expansion is expected to possibly be available for fuel storage in the early 1980's (Reference 6-41, page 4). On April 30, 1977, an application for a license to expand the present Morris facility was sent to the NRC (Reference 6-41, page 4). The application describes a new, or third, basin 92 feet long, 40 feet wide, and 29.5 feet deep (water depth of 28.5 feet) (Reference 6-16, page 3-4). The width is the same as the existing pools and will utilize the same fuel movement crane and fuel storage system as is used in Basins 1 and 2 (Figure 6-13). The storage system will consist of stainless steel fuel baskets locked in a stainless steel support grid on the basin floor. The proposed basin will eventually hold about

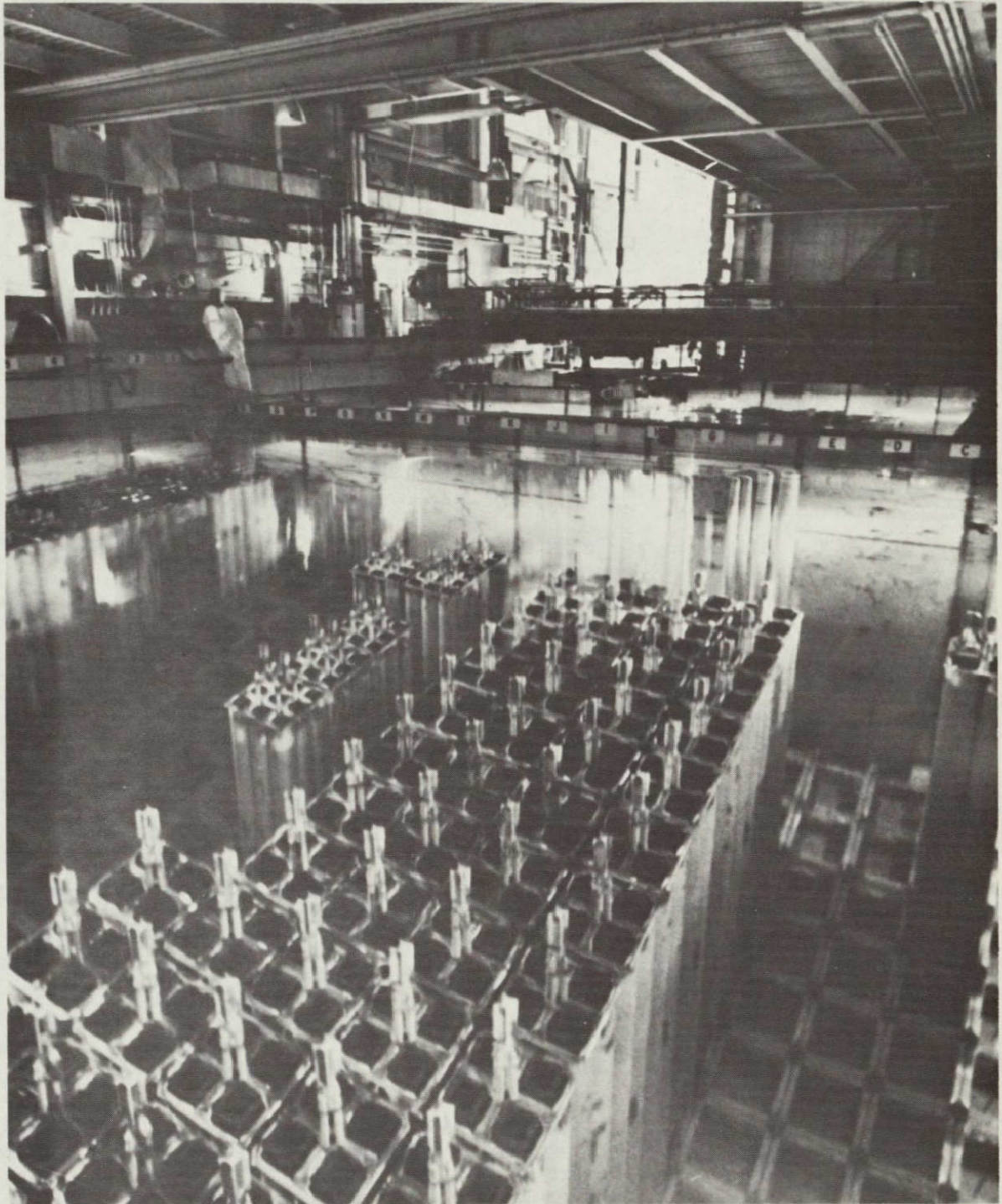


Figure 6-13. Fuel Storage Basin Areas: Viewed from the south wall of Basin 2, looking across both fuel basins towards the cask unloading basin and the decontamination area. The tall doors to the receiving area are open. In Basin 2 (foreground) both PWR baskets (4 bundles per basket) and BWR baskets (9 bundles per basket) are in place in the support grid. The bridge of the basin crane is in the background. (Ref. 6-15) (Photo Courtesy of General Electric)

659 fuel baskets (around 1/3 PWR bundles and 2/3 BWR) representing 1100 MTU in the stored fuel. (Reference 6-16, page 3-1, 6-40, page 9).

A new water cleanup system for Basin 3 will be provided. The water will be cooled by operation of the presently idle cooling banks. (Reference 6-16, page 3-6, 3-7). (Only one of the three cooling banks is required for Basins 1 and 2; the remaining two banks are standby). (Reference 6-16, page 3-6).

An air conditioning system will be installed in the existing ventilation system to control temperature and reduce humidity in the basin area. (Reference 6-16, page 3-8).

The building structures covering the basin and associated equipment will be similar to the existing structures. (Reference 6-16, page 3-1 to 3-11). (Figure 6-13.)

6.2.4 Issues--Engineering Achievability

The primary function of an independent spent fuel storage facility is to receive, store, and eventually ship irradiated fuel with minimal impact on the health and safety of the public--under both normal and abnormal (accident) operating conditions. The specific guidelines for the protection of the public from excessive exposure to ionizing radiation are set down in Parts 20 (for normal operating conditions) and 100 (for accident conditions), Title 10, of the Code of Federal Regulations. With respect to accidental releases of radioactive material the federal regulations (Part 100) indicate that an individual's whole body total radiation dose is not to exceed 25 rems, and that the dose to the thyroid from iodine exposure is not to exceed 300 rems. These upper guidelines are related to the dose a person located at any point on or beyond the site boundary could receive from the entire passage of a radioactive cloud. The cloud would be the result of a postulated fission product release at the storage facility.

There are three principal sources of potential radiation hazard to the public.

- (1) Direct radiation from the irradiated fuel
- (2) Liquid release to the environs from either an excessive storage basin leak or low activity waste vault leak
- (3) Airborne release from a potential fuel cladding rupture or cooler leak

An event diagram for postulated accidents is shown in Figure 6-14.

The following accident safety analysis is based on information concerning the GE Morris Operation. It is a summary of the analysis of postulated accident events as utilized by GE to minimize the cause of such events, quantitatively identify and mitigate the consequences, and evaluate the ability of the Morris Operation organization to cope with each situation

should it occur. The information base for this analysis was obtained from in-house GE research efforts as well as from numerous other research organization (such as Dames and Moore, H. J. Sexton & Associates, Battelle Pacific Northwest Laboratories and the Programmed & Remote Systems Corporation) (References 6-43 thru 6-50). The various component and systems safety analyses which have been published in numerous documents, reports, and license applications were brought together and presented in one major document - the Consolidated Safety Analysis Report. (Reference 6-15) It is from this report that the following material was obtained but representing the efforts of numerous people and organizations over a number of years. Such analyses of the operating Morris facility is representative of similar efforts undertaken for the AGNS storage facility and that which will be done for proposed independent, or central, spent fuel storage systems (such as at the proposed Exxon reprocessing plant).

6.2.4.1 Direct Radiation. Direct radiation can occur only when the water level in the basin is too low to provide adequate shielding. This can be the case when there is either excessive evaporation (a cooling system failure) or excessive basin leakage. These failures must occur in conjunction with an inadequate makeup water capability to pose a potential radiation hazard situation. Theoretically, a basket could be dropped in such a manner that the fuel could spill out and come to rest in a critical array. Radiation exposure to those persons in the basin area could possibly result due to water level being inadequate in depth to provide shielding from such a critical array. (See the fuel drop and missile impact sections for descriptions of the events required to occur before criticality could result.)

(1) Excessive Evaporation--Loss of Fuel Basin Cooling. (Reference 6-15, page 8-4 to 8-5) The existing system for cooling the storage basin water at the Morris facility consists of three separate cooling banks. The operating bank has a heat removal capacity of 1.76 megawatts of heat (6×10^6 Btu/hr) while the two standby banks have a total capacity of 2.93 megawatts (10×10^6 Btu/hr). (Reference 6-15, page 1-39) These two idle banks will be placed in service as required by the increased heat load as fuel is moved into the basins. The cooling capacity is considered by GE to be adequate to handle either the present two basin maximum heat load of about 1.9 megawatts (6.5×10^6 Btu/hr), or the proposed three basin maximum of 3.98 megawatts (13.6×10^6 Btu/hr). (References 6-16, page 3-6 and 6-15, page 5-39)

The planned basin expansion at Morris will include the installation of a new pump and associated valves and piping with provisions to add another pump if required. Currently there are two 750 gpm pumps at the facility. The temperature of the basin water at the Morris facility is maintained between 25-30°C (78-86°F) (Reference 6-41, page 6) The Morris Operation cooler is not always operating as there is a significant amount of natural heat dissipation by the pool structure and atmosphere. Most of the decay heat generated by the fuel is removed by evaporation and the remainder is conducted through the concrete walls. This conduction and evaporation is generally adequate to maintain the water temperature at less than 44°C (111.2°F) without the cooling system in operation providing there is sufficient cool makeup water. (Reference 6-15, page 5-39)

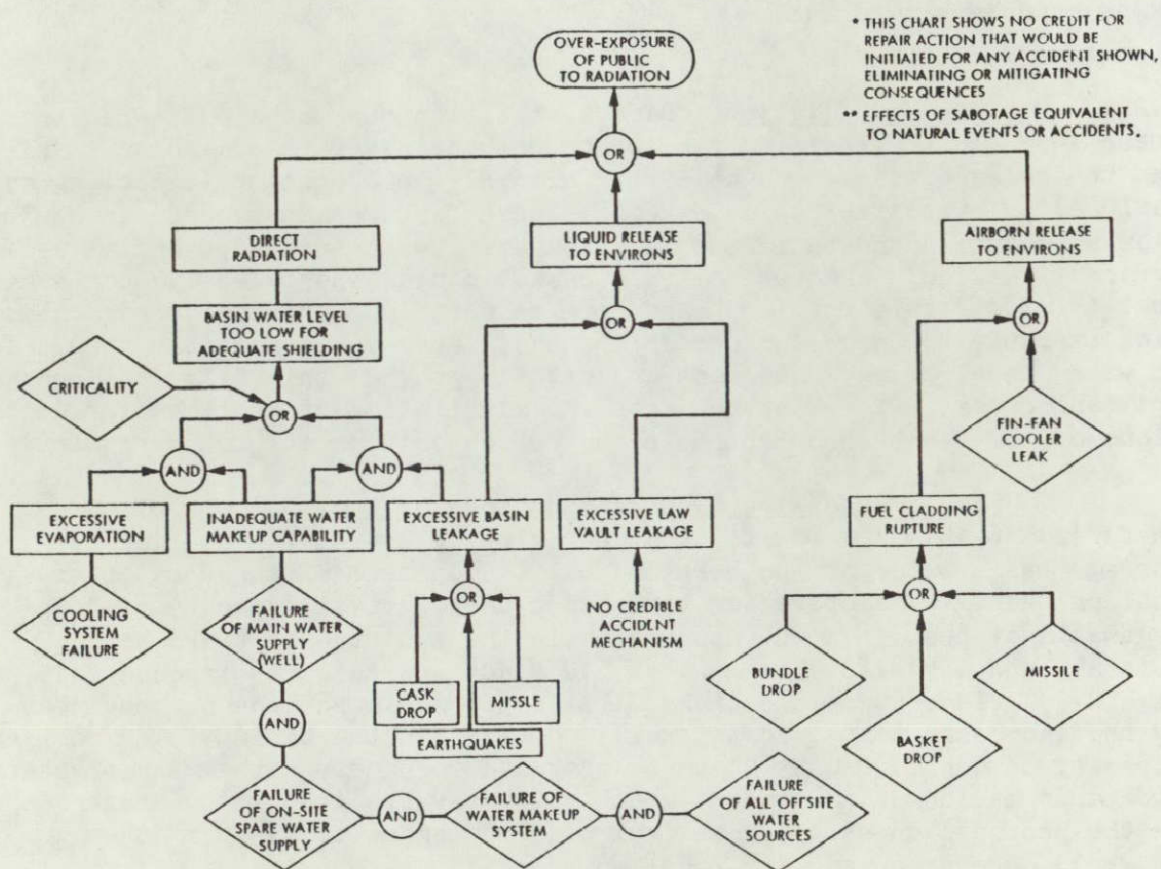


Figure 6-14. Event Diagram for Postulated Accidents
(Ref. 6-15, pg. 8-3)

The situation where the basin water could fall too low for adequate shielding must involve the failure of the cooling system and of the makeup water supply system to provide sufficient water to compensate for evaporative losses. For the case where both basins are filled with spent fuel and assuming the resulting heat load is 1.87 megawatts (6.4×10^6 Btu/hr) (the maximum two basin heat load), it has been calculated that it would require at least nine days to evaporate the more than 39,000 ft³ of water before the tops of the fuel would be exposed. (Reference 6-15, page 8-5) In reaching the boiling point the water temperature would rise less than 2°F/hr for about three days and more slowly thereafter (Reference 6-18, pg. 25). Water level indicators and alarms would notify operating personnel as soon as the water level began to decrease and corrective action could then be taken before any major water loss occurred.

(2) Inadequate Water Makeup Capability. The primary source of water for the Morris facility is a deep well (788 feet) located on the site. (Reference 6-15, page 2-2) A 100 gpm submersible pump provides water principally for sanitation purposes, with some water (up to 1400 gallons per day) for basin evaporation and utility boiler water makeup. (Reference 6-15, page 3-40) A backup well (383 feet deep) is located on the Morris site about 1/4 mile to the east of the storage facility. Provisions have been made in the event the situation requires it, to obtain makeup water from the nearby (3/4 mile) Dresden Nuclear Power Stations. (Reference 6-15, page 2-2) Also water can be obtained from the Kankakee River. As a result of these multiple sources of water, there is a very small probability that there would not be sufficient makeup water available in the event of a reduction of storage basin water level before the spent fuel was inadequately shielded.

(3) Excessive Basin Leakage--Cask Drop, Missiles, Earthquakes. (Reference 8-11 to 8-15) There are no piping penetrations which, if open, could drain the spent fuel storage basin. There are also no potential siphoning pathways from the basin. Inadvertent drainage of the basin must therefore result from the penetration of the liner.

The fuel unloading basin at the Morris facility is set directly against bedrock. (Reference 6-15, page 5-33 to 5-36) The walls and floor are reinforced concrete lined with stainless steel plates. The steel liner sheets, as seen in Figure 6-14 being installed in Basin 2 are welded together and placed flush against the concrete wall. Drain slots are formed in the concrete walls and floor to facilitate drainage of any water that may collect between the steel liner and the concrete. These 1/2 inch by 1/2 inch drain slots, as seen in the top of Figure 6-14A, lead to a collection header at the base of the walls which drains the water into a sump (a six-inch pipe which extends from the top of the wall to a foot below the unloading pit floor). Liquid level detection monitors as well as a pump-out system is located in the sump.

In June, 1972, an IF-100 shipping cask tipped against the wall of the unloading pit. The pit liner was ruptured with the result that basin water entering the space between the liner and the structural concrete wall. (Reference 6-51, page 7-8, and a more detailed description is found in Reference 6-15, page 8-5 to 8-11) Only minor damage was sustained by the cask. A temporary patch was installed on the liner within 27 hours and a permanent repair was made in 12 days after the incident. There was some

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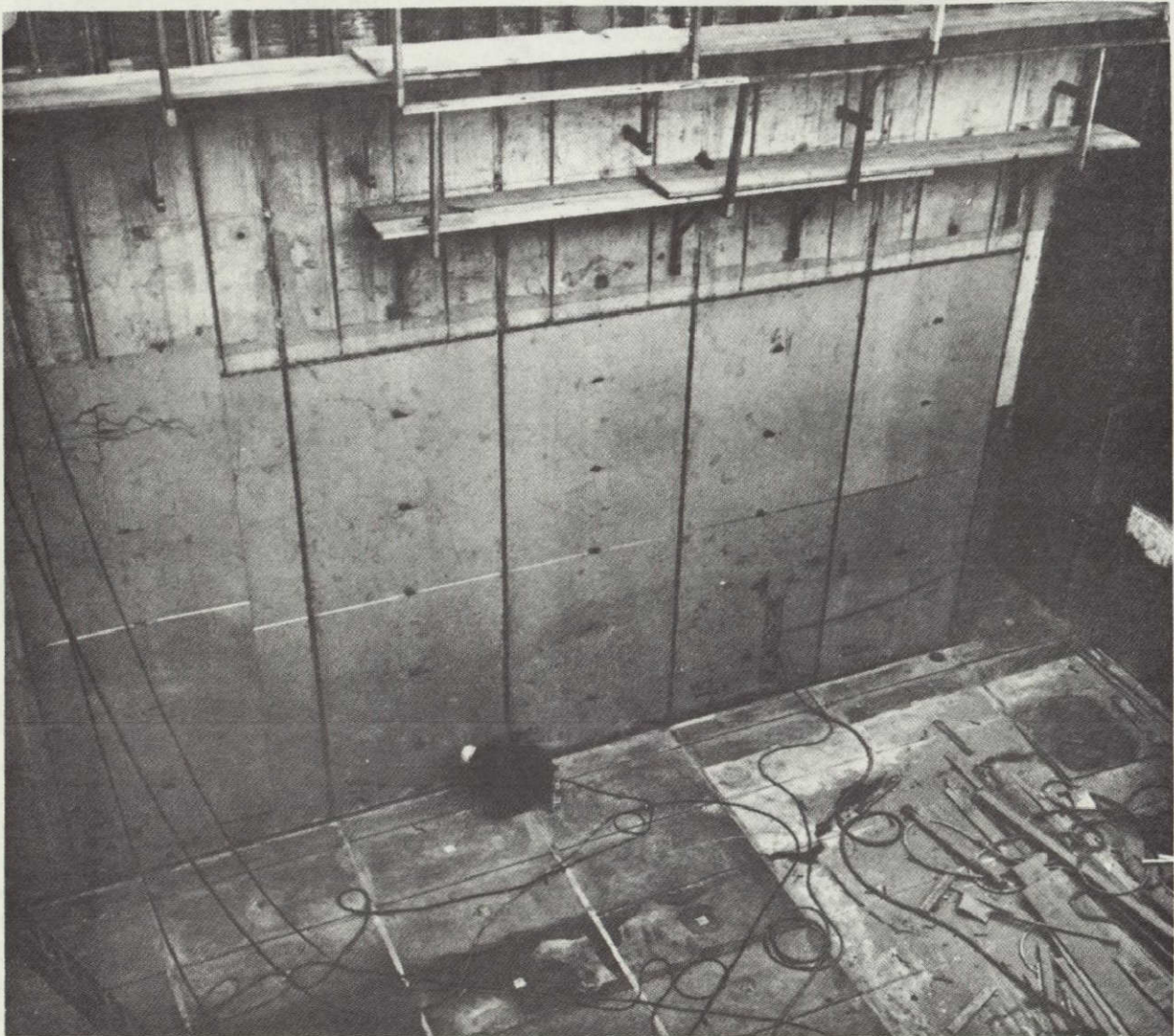


Figure 6-14A. Stainless Steel Basin Liners: Both storage basins, and the unloading basin are completely lined with stainless steel sheets (304L) placed flush against the concrete walls and floor, tack welded to a grid of stainless steel back-up members (view facing north wall of Basin 2). (Reference 6-15)

seeping of water into portions of the canyon and main process building. A total of approximately 2500 gallons of water containing an estimated 6 mCi of activity (primarily Cs-134 and 137) was not recovered. (Reference 6-15, page 8-9). It was assumed that most, if not all, the water remained within the confines of the structure and was contained in minute fissures in the concrete around the fuel and waste storage basin and process cells. The result of the leak was concluded by the AEC Staff to be that any radioactivity that may have been released as a result of the incident had no significant impact on the environment. (Reference 6-51, page 8) No groundwater contamination above background levels could be detected (Reference 6-15, page 8-5). As a result of this incident numerous operational, technical, and environmental monitoring steps were taken to prevent and mitigate any such future accident. (Reference 6-15, page 8-11, 8-10)

For future storage pools, both in conjunction with the reactor and independent of the reactor, the liners are expected to be prefabricated in large sections at a central fabrication facility and assembled on the construction site. The drain channels will be welded directly to the back of the liner as opposed to being formed in the concrete wall. An example of such a pool liner is seen in Figure 6-14B. This liner, though for new fuel storage at a reactor site (constructed by the Boeing Company for WNP-1), is representative of such a prefabrication operation. Ease of fabrication and quality control are two of the major benefits of factory versus on-site fabrication. Other advantages are reduced labor cost and decreased interference with other pool construction operations.

At the Morris facility the floor of the unloading basin shelf and deep pit include provisions to dissipate impact loads from a maximum cask drop accident. The shelf has a fabricated, stainless steel crushable finned pad and a 2-inch steel plate on top of the stainless steel liner. The floor of the pit is covered by a 1 3/4-inch steel plate under the stainless steel liner to dissipate the impact load from a 70-ton shipping cask (IF-300) falling 31 feet. (Reference 6-15, page A. 7-1 to A. 7-8)

The shipping cask is first lowered onto the shelf (18.5 feet below the water surface) where the rigging is changed. The cask is then moved over the deep pit and lowered to the floor (another 29 feet). In this manner the maximum postulated cask drop is 31 feet. The analysis of a cask drop considered the fall of an IF-300 cask (the largest shipping container to be handled at Morris within the immediate future) in such a manner as to allow minimum energy absorption by the cask fins and therefore the highest load on the floor.

The strength of the cask fins will result in the total energy of the drop being absorbed by the pad. The finned pad on the unloading shelf (two 1-inch stainless steel plates separated by 4 inch high, 1/2 inch thick steel fins) is expected to absorb all of the cask's energy (with one half the total drop energy being accounted for by force deflection curves and the other one half being absorbed by the bending of the pad's fins) (Reference 6-15, page A.7-1 to A.7-8 and A.13-2 to A.13-9). For a drop on the shelf edge or pit floor, the load on the shelf concrete is less than its ultimate dynamic load and from the analysis of the loads and materials

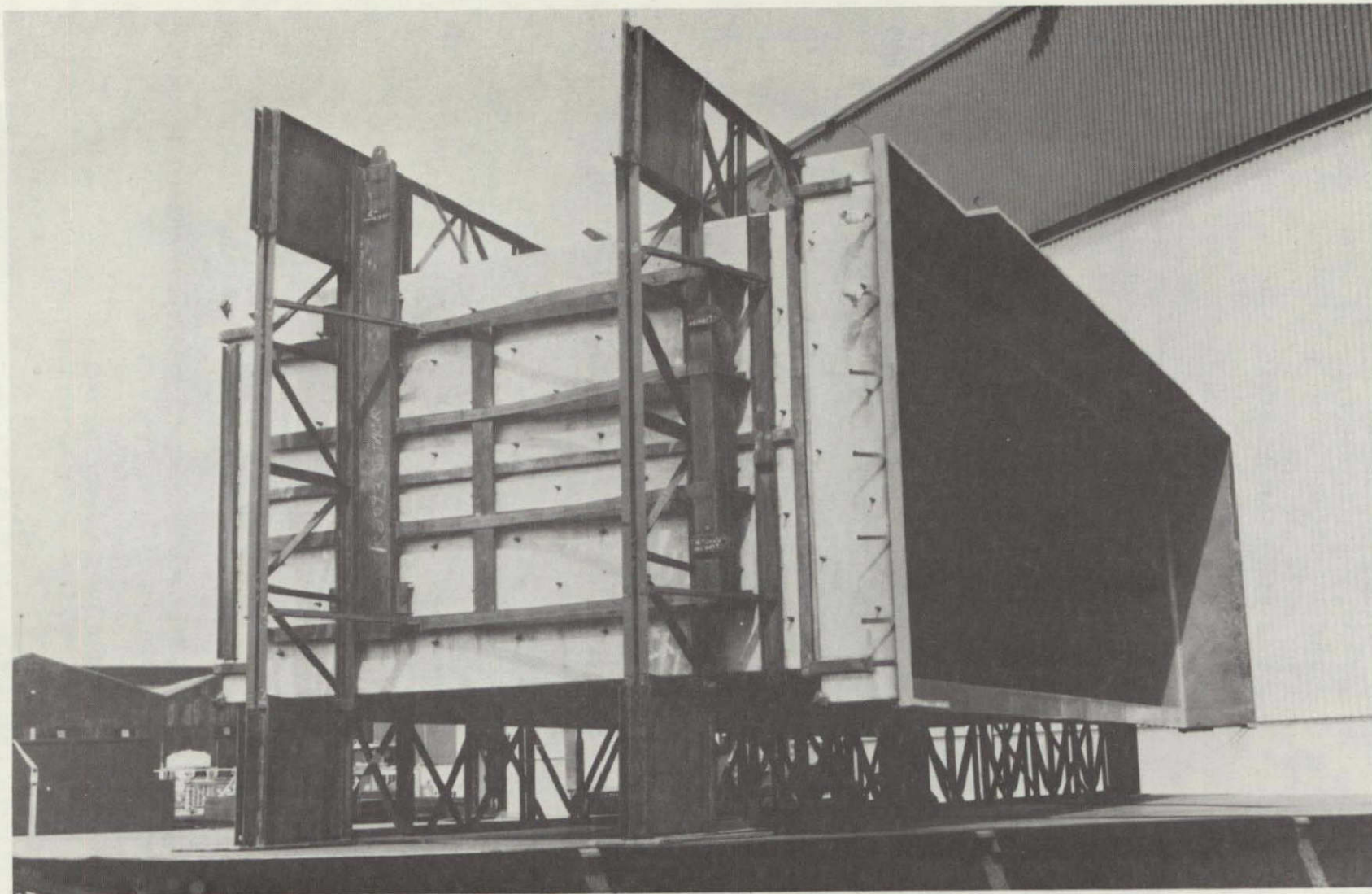


Figure 6-14B

Stainless steel pool liner for the new fuel storage pool at the WNP-1 reactor. Dimensions are 17 feet deep, 34 feet long, and 10.5 feet wide. The drain channel can be seen behind the center weld running from the top collection header to the lower drain header. The exterior structure is for shipping strength. (Photo courtesy of Boeing, Seattle, WA)

strengths the integrity is expected to be protected due to the distribution of the fallen load forces over a relatively large area. (Reference 6-15, page A.73-9 to 13-12)

If the concrete were to fail, penetration is expected to be less than an inch and the stresses on a flat plate that is deflected an inch is well within the elastic stress range for steel. The conclusions are that the floor and shelf of the unloading pit are adequately protected from a postulated IF-300 cask fall of 31 feet from the shelf to the floor or 18.5 feet to the basin shelf from its highest movement point. (Reference 6-15, page A.7-1 to A.7-9, A.13-1 to A.13-12)

The facility structures and components essential for safety are designed to withstand the effects of wind borne missiles without the loss of the capability to retain the radioactive inventory. In the analysis of a postulated airborne missile striking the fuel basin liner, a 300 mph maximum velocity tornado was assumed. An examination of the various objects which could potentially damage the basin revealed that only such objects in the general vicinity of the basin which were too heavy to be lifted but could be accelerated by tornado winds and deflected or dropped into the basin could pose any concern. (Reference 6-15, page 4-7 to 4-18) The analysis determined that a small (1800 pound) car being deflected into the basin at a maximum horizontal velocity of 242 ft/sec (maximum vertical velocity of 171 ft/sec) would not have sufficient energy to penetrate the 1/16 inch basin liner or the 3/16 inch basin floor due to the large impact cross section. It was also concluded by the GE analysis that a 630 pound, 12 inch diameter 20 foot long telephone pole with a maximum horizontal velocity of 264 ft/sec (maximum vertical velocity of 187 ft/sec) may deform the liner but could not penetrate the wall or floor liner of the fuel storage basin. (Reference 6-15, page 4-9 to 4-19, 8-13 to 8-14)

6.2.4.2 Liquid Release to Environs

(1) Excessive Low Activity Vault Leakage. (Reference 6-15, page 8-15) The low activity vault is a reinforced concrete structure with an internal holding tank. The vault is designed to withstand a design basis earthquake. Being underground, the structure is protected from the potential effects of a tornado or tornado generated missile. Also no heavy objects are moved over the vault. As a result of the design of the vault, GE has concluded that there is no credible accident that could rupture the tank and the concrete sufficiently to release liquid radioactive material to the environs. (Reference 6-15, 8-15) The structure is equipped with a leak detection system and pumpout system to handle minor leaks.

6.2.4.3 Airborne Release to the Environs. (Reference 6-15, page 8-15 to 8-21) The two potential methods for release of radioactive gases are the ruptures of the fuel cladding and the leakage of the cooling system heat exchanger. The fuel cladding may be ruptured by either dropping the fuel bundle, dropping a loaded fuel basket, or by having a missile strike the stored fuel.

In the safety analysis for the Morris spent fuel facility, a number of basic assumptions were made concerning the three potential cladding rupture events. The fuel is considered to have a burn up of 44,000 MWD/MTU and to have been cooled 90 days before storage in the Morris basin. Because of the negligible particulate activity available for release, no solid fission products are considered released. The ventilation air flow rate from over the basin to the environment via the main stack (100 m) is assumed to be 6,500 cfm. The overall basin water decontamination factor (ability of the basin water to retain certain elements) for iodine is considered conservative at 500. A factor of only 1 is used for noble gas due to water's negligible effect on their removal. Also the worst case atmospheric dispersion factor is considered to be 2.8×10^{-5} sec/m³. (Reference 6-15, page 8-20 to 8-21)

Of the fission gases which accumulate in the fuel during irradiation, krypton-85 is the most abundant by the end of 180 days of cooling. The other long-lived radioisotope in the fuel is Iodine-129. Xenon-131m, Xenon-133, and Iodine-131 all decay relatively rapidly.

The amount of radioactive gas released from the fuel is dependent on the fuel temperature. Release fractions, or percentage of gas produced during irradiation that is released from the fuel pellets, ranges from 20 to 45% depending on the irradiation history. For fuel with a burnup of 44,000 MWd/MTU and 90 days of cooling the percentages used in the cladding/rupture analysis at the Morris facility were: (Ref. 6-15, pg. 8-19)

Kr-85	30% of total inventory in fuel gaps
Xe-131m	10%
Xe-133	10%
I-131	2%
I-129	2%

Bundle Drop. (Reference 6-15, page 8-21 to 8-24) The accidental drop of a fuel bundle which could result in mechanical damage to the fuel and the subsequent release of fission products would most likely occur in the unloading basin. The fuel is assumed to be dropped from the maximum height it would be lifted during transfer. Although it is a low probability event, all the fuel rods in the bundle are assumed to rupture, releasing the fission gases to the basin water. Based on the movement of air over the water basin and through the stack (6500 cfm), and assumption that all released fission gases are expelled from the stack, and that the duration of release is 2 hours, the resulting radiation doses at the site boundary were calculated to be: (Ref. 6-15, pg. 8-23)

<u>Body Organ</u>	<u>BWR</u>	<u>PWR</u>
Whole Body	9×10^{-3} mrem	1.9×10^{-2} mrem
Thyroid	4.4×10^{-2} mrem	9.6×10^{-2} mrem

Basket Drop. (Reference 6-15, page 8-24 to 8-26) The maximum height a fuel basket would be above the floor of the fuel storage basin would be 3 feet. In the unloading basin the basket would reach a height of 22.5 feet above the basin floor (equivalent to a 12.6 foot air drop distance). If the basket fell from this height the basin liner, the basket, and the fuel could be damaged. To prevent the spillage of the fuel from the basket in such a fall a guard structure is situated in front of the entrance to the spent fuel basin. The guard structure holds the basket in a vertical orientation in the event of an accident.

Although the fuel is not likely to be severely damaged, for the safety analysis it was assumed all the fuel rods in all the bundles in the basket are ruptured and released all the fission gases in the rod gaps. Again the duration of the release was 2 hours. The maximum offsite exposure results were calculated to be: (Ref. 6-15, pg. 8-26)

<u>Body Organ</u>	<u>BWR</u>	<u>PWR</u>
Whole Body	8.1×10^{-1} mrem	7.84×10^{-1} mrem
Thyroid	3.96×10^{-1} mrem	3.84×10^{-1} mrem

Cooler Leak. (Reference 6-15, page 8-31 to 8-32) If the coils of the cooler (heat exchanger) leaked, much (if not all), of the water would be evaporated into the air stream which is blown across the coils. A 1.5 gpm leak is assumed for the safety analysis. A leak of this size would result in a visible vapor plume. The radioactive material concentration in the water flowing through the coils is generally around 6×10^{-4} Ci/ml. For the GE analysis all the activity was assumed to be Cs-134. The maximum dose rate at the site boundary was calculated to be 1×10^{-3} mrem/hour whole body and 7×10^{-4} mrem/hr to the thyroid (Ref. 6-15, pg. 8-30).

The exposure levels at the site boundary as described in the previous sections for various postulated incidents are only a fraction of the Federal guidelines values. Also these exposure levels were determined on the assumption that no corrective steps were taken and that there was no air filtration. Alarms for the decrease in the water level and airborne radiation would notify the operating personnel as soon as an incident occurred and corrective measures could then be taken. The result would be a very minor (a small fraction of the previously mentioned exposure levels) or non-detectable level of radiation reaching the site boundaries from the major events as postulated in the previous sections.

6.3 ENGINEERED INTERIM SURFACE OR SHALLOW SUB-SURFACE STORAGE

A interim alternative to the placement of spent fuel in a geological repository (discussed in Section 6.4) is to take the unpackaged spent fuel assemblies from either the reactor site or a centralized storage basin to an engineered spent unprocessed fuel facility (SURFF). This facility would place the fuel assemblies in a metal container to isolate the fuel rod cladding from the external environment, and

would store them retrievably in an area that would provide biological shielding from the decay heat and radiation produced by the spent fuel.

The use of an engineered surface storage facility for spent fuel has only recently been suggested as an option by the ERDA (Reference 6-24 pg. 3; and Reference 6-25, pg. 21). However, it is not a new idea. Much of the design and development work was initiated in the early 1970s during the development of a concept for the surface storage of high-level waste from reprocessed fuel--the retrievable surface storage facility (RSSF) (Reference 9).

It should be kept in mind that currently there are several design concepts under consideration, thus this section discusses a set of concepts. The packaging and surface storage of spent fuel is discussed as one component because of the present lack of design specificity and because of the close interrelationship of the technical problems associated with packaging and surface storage.

At this time there are no performance criteria for a SURFF or its related packaging facilities, but the existence of such criteria would be unexpected given that the concept has only recently been seriously proposed.

6.3.1 Process Description

As noted previously, a spent unprocessed fuel facility (SURFF) is a broad concept with many design alternatives. These alternatives can be divided into two classifications: (1) unpackaged storage and (2) packaged storage, as shown in Table 6-1. The former is considered in the TAD documents to be appropriate only for storage for a 10 to 20 year period; otherwise packaged storage is suggested (Reference 6-21, page 17.1). Although it is not yet clear how long a SURFF should or will operate, the latter classification appears to provide more flexibility in storage strategy without a considerable cost impact. Whether or not the packaging will cause complications during reprocessing (if reprocessing is allowed) needs to be examined. Another factor is that a passive or natural cooling system is preferred over a forced cooling system in the SURFF concepts under consideration. It is the emphasis on passive system that most clearly differentiates SURFF from central pool storage.

Table 6-1. Summary Comparison of Spent Fuel Storage Alternatives

<u>Storage Alternative</u>	<u>Confinement Barriers in Addition to Cladding</u>	<u>Means of Heat Removal</u>	<u>Method of Controlling Fuel Cladding Corrosion</u>	<u>Maintenance Requirements</u>	<u>Land Use (Surface)</u>
Unpacked Storage					
Water basin	Filters	Forced circulation of basin water	Water quality control	High	Low
Air-cooled vault	Filters	Forced circulation of air	Low temperature	Moderate	Moderate
Packaged Storage					
Water basin	Filters & package	Forced circulation of basin water	Packaged in inert or noncorrosive medium	High	Low
Air-cooled vault*	Package	Natural circulation of air	Packaged in inert or noncorrosive medium	Low	Moderate
Concrete surface* silo	Package	Natural circulation of or conduction to air	Packaged in inert or noncorrosive medium	Low	High ^(a)
Geologic	Package, hole liner & filters	Conduction to earth	Packaged in inert or noncorrosive medium	Moderate	Low
Near-surface* heat sink	Package, hole liner	Conduction to earth	Packaged in inert or noncorrosive medium	Low	High

a. Estimated to be ~120,000 m² per 1500 MT, assuming one MT per storage unit.

* These three are the major SURFF alternatives presently being evaluated

Reference 3, page 17.2

6.3.1.1 Packaging. The use of multiple barriers for containment is common to all of the design alternatives being studied by ERDA. The first barrier for all modes of spent fuel storage is the fuel cladding.

"The cladding withstands a far more-severe environment in the reactor than it would encounter in a storage facility, although for a shorter period of time. In concept, a storage facility is designed to protect the fuel cladding against mechanical, chemical, or or thermal damage." (Reference 6-21, page 17.1)

The second barrier is provided by sealing the fuel assembly inside a container. This process could take place three to four years after the fuel has been discharged to allow for a decrease in heat generation and radiation. Preliminary cost analyses indicate that the sealing or packaging would take place at the SURFF and not at the originating reactors. Although such cost analyses were done in reference to RSSF (Reference 6-19) the logic appears to still hold true for SURFF. This procedure reduces the capital investment by not requiring packaging facilities at each reactor. In addition, increased transportation costs attributable to the increased weight of the container are also avoided.

The centralized packaging facility must carry out the following functions:

- receive and handle spent fuel (see description of the existing facilities at the Midwest Reprocessing Facility in the previous section)
- package spent fuel in a manner consistent with the storage concept
- overpackage the fuel bundles or spent fuel packages that have failed in storage
- transfer packaged fuel to storage

Over the last decade, packaged spent fuel from various reactors has been stored at the Savannah River Plant (Reference 6-21, page 17.17), and at the Whiteshell Nuclear Research Establishment, Atomic Energy of Canada, Limited, efforts have begun to package and store spent fuel in facilities similar to SURFF. However, for the SURFF, as stated under 6.3, many design considerations of the fuel bundle package have not been specified. For the RSSF designs, the metal cannister in which the solid waste is placed would serve as the sole extent of packaging before the HLW was placed in storage (Reference 6-31, page 392).

The Technical Alternatives Document (Reference 6-21) describes a simple container for an intact BWR or PWR spent fuel bundle. The selected container material would probably be low carbon steel. Low carbon steel is more economical than stainless steel or any other highly corrosion resistant material, and highly corrosion-resistant materials would not be necessary except in a water basin storage concept. Present indications

are that the water basin concept with its nonpassive cooling concept will not be considered seriously as a SURFF concept (Reference 6-27). "Low carbon steel corrodes about 0.0005 cm/yr at 260°C when shielded from weather" (Reference 6-21, page 17.19). Therefore, a proposed wall thickness of 1.3 cm would be good for at least 100 years. The specific design thicknesses given below are dependent upon the SURFF concept utilized.

To transfer the heat generated by the spent fuel to the container walls, a material with high thermal conductivity would be used to fill the space between the fuel rods. Helium is most frequently considered (Reference 6-21, page 17.19). Other possible transfer mediums are zinc, powdered aluminum in a helium atmosphere, and air. The advantages and disadvantages of each of these approaches need to be compared.

The Technical Alternatives Document notes that "additional development is required on packaging concepts for long-term storage of spent fuel". However, no technological barriers are anticipated. Two to four years would be required prior to the final design of such a facility (Reference 6-21, page 17.17)

6.3.1.2 The Storage Facility. Discussions with ERDA indicate that there are three passive systems under consideration for a SURFF:

- Sealed Storage Cask Concept (SSCC)
- Air-cooled Vault Concept (ACVC)
- Storage in Near-surface Heat Sink Source (Reference 6-27)

The first two, in addition to storage in a water basin, were originally suggested in WASH-1539 as the RSSF concepts. The Committee on Radioactive Waste Management (CRWM) of the National Academy of Science-National Research Council, in its review of the RSSF concepts, recommended that an optimized version of the Sealed Storage Cask Concept (SSCC) be pursued (Reference 6-19, page 2). The present effort in Canada for the storage of CANDU fuel is stressing the use of this alternative (Reference 6-21, page 17.41).

The SSCC calls for "spent fuel assemblies packaged in 5-cm-thick, low carbon steel containers, placed in concrete silos which serve as neutron shields, and (are) stored vertically on outdoor concrete pads" (Reference 6-21, page 17.35). The SSCC, like SURFF itself, is really a conceptual design with various alternatives. Three specific concepts were described in the RSSF Engineering Studies, and comparison of the design specifications is shown in Table 6-2 (Reference 6-20, page 1-2). The NAS review of RSSF found the SSCC-Shielded to be the "forerunner" of the SSCC concepts. The SSCC-Thick Wall was considered to be "less than optimum with respect to shielding efficiency and use of materials, while the SSCC-Unshielded did not provide adequate biological shielding" (Reference 6-19, pages 22 and 23). The Technical Alternatives Document (TAD) describes only the use of a 5 cm thick package for spent fuel storage. It is suggested that for cost efficiency, packaging of multiple assemblies

Table 6-2. Comparison of Retrievable Surface-Storage Facility Concepts

	Sealed Storage-Cask Concepts			Air-Cooled Vault Concept (ACVC)	Water Basin Concepts	
	SSCC-Shielded	SSCC-Thick Wall	SSCC-Unshielded		WBC	WBC-OP
Cooling method	Air natural circulation	Air natural circulation	Air natural circulation	Air natural circulation and cooling	Water-forced circulation and cooling towers	Water-forced circulation and cooling towers
Confinement method	2" Wall sealed carbon-steel cask welded closure	14" Wall sealed carbon-steel cask welded closure	8" Wall sealed carbon-steel cask welded closure	0.5" Wall carbon steel overpack welded closure	Water-filled steel-lined concrete basin	0.5" Wall, stainless steel overpack welded closure water-filled basin
Biological shielding	38" Thick concrete sleeve	11" Thick concrete sleeve	None (distance)	Concrete vaults	Water basin	Water basin
Storage concept	Vertical outdoor on 20-foot centers	Vertical outdoor on 20-foot centers	Horizontal outdoor 20 ft apart	Vertical indoor	Vertical indoor	Vertical indoor
Maximum normal operating temperature (°F)						
Canister (outer surface)	470 ^a	450 ^a	455 ^a	620 ^b	<128	484
Overpack (outer surface)	245 ^a	170 ^a	180 ^a	400 ^b	NA	127
Concrete (inner surface)	155 ^a	122 ^a	NA	200 ^b	NA	NA
Dose rates (mrem/h)						
Outside surface	2(concrete sleeve)	2(concrete sleeve)	14,000	<2(top of deck)	<1 (water surface)	<1 (water surface)
Within array	<4	<4	10,000	NA	NA	NA
Resources (to year 2010)						
Land(acres)	1,100	1,100	2,300	100	100	150
Water(gal/yr)	75 x 10 ⁶	75 x 10 ⁶	75 x 10 ⁶	75 x 10 ⁶	1,500 x 10 ⁶	1,500 x 10 ⁶
Carbon steel (tons)	0.28 x 10 ⁶	2.3 x 10 ⁶	0.98 x 10 ⁶	0.15 x 10 ⁶	0.10 x 10 ⁶	0.22 x 10 ⁶
Stainless steel (tons)	42	42	42	42	5,600	45,000
Concrete(yd ³)	2.7 x 10 ⁶	1 x 10 ⁶	0.09 x 10 ⁶	0.34 x 10 ⁶	0.1 x 10 ⁶	0.14 x 10 ⁶
Personnel	330	330	330	280	250	330
Power(kWe)	2,500	3,000	3,000	2,100	10,400	10,000
Capital ^d (in 1973 dollars)	900 x 10 ⁶	2,600 x 10 ⁶	1,130 x 10 ⁶	460 x 10 ⁶	280 x 10 ⁶	740 x 10 ⁶
Cumulative operating costs(in 1973 dollars through year 2010)	220 x 10 ⁶	200 x 10 ⁶	190 x 10 ⁶	164 x 10 ⁶	170 x 10 ⁶	220 x 10 ⁶

SOURCE: *Retrieval Surface Storage Facility: Alternative Concept Engineering Studies*, ARH-2888, Rev., Atlantic Richfield Hanford Company, Kaiser Engineers (July 1974).

^a80°F ambient air temperature.

^b110°F ambient air temperature.

^cIncluding structural and reinforcing steel.

^dIncluding cost of steel overpacks and concrete shields

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should be used to minimize the required number of casks or silos. Cost efficiency considerations are dependent on the cumulative discharge of spent fuel. Therefore it is useful to note that the RSSF was designed to handle approximately 80,000 cannisters, which was expected to be all of the high-level wastes generated through the year 2000. Present projections by ERDA of operating plants indicate that by 1986, 96,000 fuel assemblies will have been discharged. A low case projection through the year 2000 shows a cumulative discharge of 320,000 assemblies (See paragraph 4.2.2). If 4 bundles could be stored in a container, 80,000 casks would be required by the year 2000. "The natural draft-cooled concept could store up to 6 PWR or 15 BWR fuel assemblies in each cask." These packages would generate about 3 kW of decay heat in 10 years of cooling (Reference 6-21, page 17.35). (The RSSF was designed to store a cannister that generated 3.25 kW of decay heat.) (Reference 6-19, page 23). Because of the greater heat loads involved with multiple containerization, longer cooling time might be required in multiple packaging. However, it appears that there would be significant cost reductions because of decreased land use, materials requirements, and handling. Final determination of the optimal method is not possible at this point due to the lack of data.

According to the TAD, the steel container for the SSCC is to be placed in a thick (38-cm) concrete silo. The concrete acts as biological shielding and represents a third barrier (Reference 6-21, page 17.57). Criticality, heat removal, and corrosion are not seen as problems. Monitoring for loss of containment of radioactive material would be handled by area survey instruments and by observation (Reference 6-21, page 17.37). In addition, the natural cooling mechanism provided by the annulus between the carbon steel cask and the concrete shield, as shown in Figure 6-15, requires no routine maintenance. Periodic inspection of the annulus and the monitoring instruments, however, would still be required. If a problem, such as clogging of the annulus, was discovered, the cask would be brought back to the packaging facility to make the required changes.

As noted, Canada is presently examining this storage concept, utilizing both conduction and convection natural cooling. Based on the RSSF design work already completed, the TAD estimated that 7 years would be required to develop an SSCC (Reference 6-21, page 17-39).

The Air-cooled Vault Concept also utilizes natural draft cooling of the spent fuel cannisters. Again, the decay heat would create the natural draft. The facility is made up of a partially buried concrete vault that is divided into cells. The canisters are stored vertically in the cells. Metal sleeves surround each storage position to help distribute the incoming cool air (Reference 6-21, page 17-31) (see Figure 6-16).

For the RSSF designs, the temperature differential would create a draft with a maximum air speed of 6 feet (1.8 m) per second can only be maintained without the use of high efficiency particulate (HEPA) and carbon filters. Such filters would be required if the first two barriers were breached. In that case, a forced cooling system would

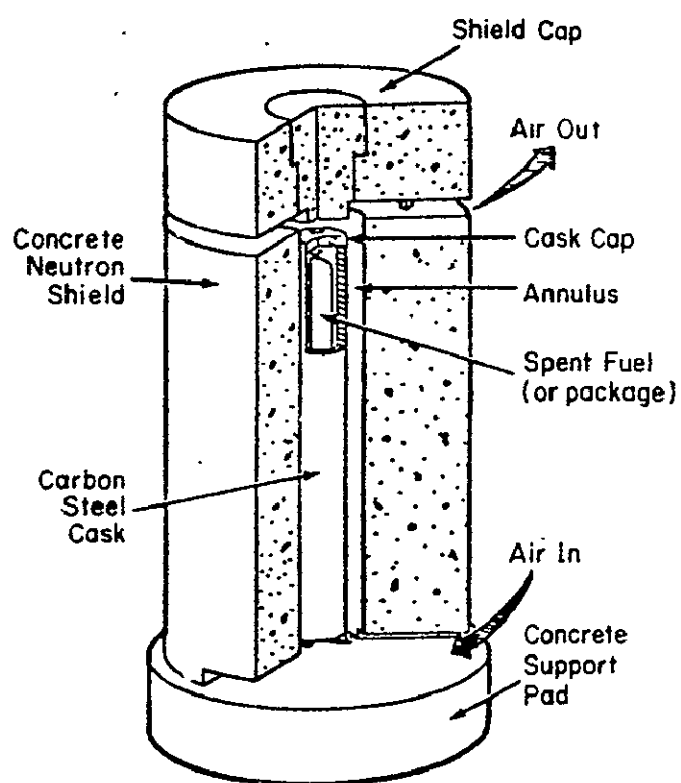


Figure 6-15. Concrete Surface Silo, Convection Cooled Concept
Source: Reference 6-21, page 17-36

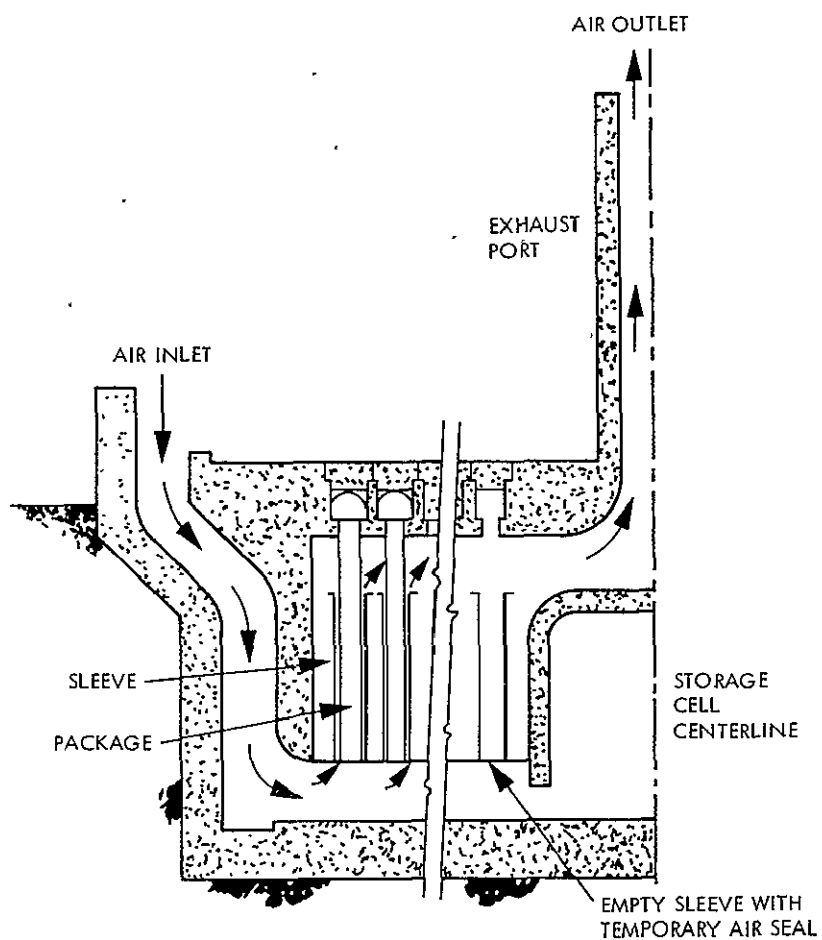


Figure 6-16. RSSF Air-Cooled Vault Concept 10
Source: Reference 6-21, page 17-31

have to be employed. Inlet and effluent air ports would require periodic inspection to ensure that the natural cooling would continue throughout the SURFF's operation.

The NAS review questioned "whether the proposed slab-shaped stack would assure a sufficiently reliable air flow to maintain temperature below design limits" (Reference 6-19, page 32). For instance, the report mentioned Taylor instabilities where local down-drafts of cool air could reduce the effective draft and thus the air flow. These problems still need to be addressed.

Presently, the spent fuel from gas-cooled reactors is being stored in forced air-cooled vaults. The spent fuel is not packaged (Reference 6-21, page 17.33). Again, based on the design work completed for RSSF, TAD estimate that design and construction of an Air-cooled Vault would require seven to nine years (Reference 6-21, page 17.33).

The near surface heat sink storage is a SURFF concept that had not been proposed for HLW storage. The concept is similar to a method used for storage of CANDU fuels (Reference 6-21, page 17.41). The canister of spent fuel is placed into concrete lined holes in the earth's surface which provide passive cooling and stable shielding. According to the Technical Alternatives Document, the concept could be "constructed and begin receiving fuel sooner than would facilities based on any other concept" (Reference 6-21, page 17-41). One outstanding question, which is an indication of the technical work needed to examine this concept, is the description given in Figure 6-17 of the packaged fuel rods. The package is said to hold 10 rods. Since 49 is the smallest number of rods in an assembly (BWR type), this requires the disassembly of each bundle prior to packaging. Whether this is an oversight in the use of the drawing or if it is based on criticality requirements has not been determined.

6.3.2 Development History

SURFF is not a new idea. Simply stated it is update of the Retrievable Surface Storage Facility (RSSF) concepts that were examined and discarded in the early 1970's. The RSSF concepts were designed for the interim storage of high-level waste. SURFF would store spent fuel bundles or assemblies. The storage of spent fuel was examined in the Technical Alternatives Document (Reference 6-21, Section 17) but most of the material appears to be based on the RSSF design concepts and the work done on designs for spent fuel storage at reprocessing plants. Present studies are now under the direction of the ERDA Richland Operation O Office (Richland, Washington). The prime contractor is Rockwell International, Inc.

ERDA is aware of the similarity of the SURFF and RSSF program. However, they are confident that the differences in the purposes of the concepts and the differences in the existing political environments will keep SURFF from being assessed in the way the RSSF was assessed, as described below. The differences as seen by ERDA are that while RSSF was an interim stop-gap facility for the storage of high-level waste, SURFF is an anti-proliferation project (Reference 6-27). It is felt that SURFF

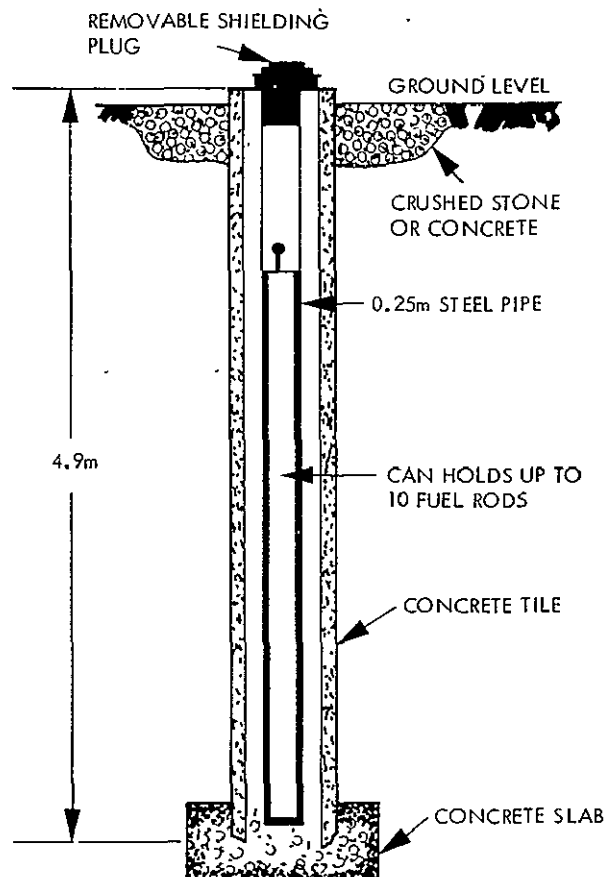


Figure 6-17. Typical Concrete Tile Hole for Surface Storage of CANDU Fuel
Source: Reference 6-21, pages 17-41

may be needed, since one of the results of the Carter policy to defer reprocessing indefinitely for anti-proliferation reasons is that the nuclear energy industry's storage capacity may be overstrained. Therefore, ERDA feels it may have to provide a technology to relieve this capacity problem until the geological disposal repositories are available.

However, ERDA still anticipates a public relations problem with SURFF because of its similarity to RSSF. Because of this similarity, it is useful to recall the brief history of RSSF.

By February 1972 the geologic repository project at Lyons, Kansas was officially terminated. While plans were made to examine novel waste management alternatives, the AEC decided it needed a short-term waste management policy. "Under the direction of the new Director of the Division of Waste Management and Transportation, Frank Pittman, the notion of an engineered Retrievable Surface Storage Facility was developed. Mausolea would be constructed in the West for the storage of AEC and commercially generated waste." (Reference 6-28, page D-14)

In September 1974, a draft environmental impact statement ("Management of Commercial High level and Transuranium-contaminated Radioactive Waste, WASH 1539) was issued. Although the scope of the study included an engineered surface storage facility for the retrievable storage of commercial high-level waste; geological formations and sites for permanent disposal as well as retrievable storage for commercial transuranium contaminated waste were also considered. Comments received from environmental groups and from State and local governments were generally critical. The Environmental Protection Agency in its unpublished comments to the AEC, stated:

"The development of an environmentally acceptable system for permanent disposal of commercially generated radioactive waste would appear to be high priority program that is essential for the development of nuclear power. However, the draft statement does not contain adequate description of a program to develop such a permanent disposal system, nor does it reflect either the priority attached to this overall program by the AEC nor an indication of the resources required. Because of the overwhelming need to develop an environmentally acceptable ultimate disposal method and the realization that there is a risk of failure in any research and development effort, we believe that work on promising alternatives should be pursued concurrently."

"A major concern--the employment of the RSSF concept-- is the possibility that economic factors could later dictate utilization of the facility as a permanent repository, contrary to the stated intent to make the RSSF interim in nature. Economic factors would consist mainly of the fiscal investment attendant to its construction and the activities which arise in the commercial segment of the economy to support its operation. Since there are controlling environmental factors that must be considered before final disposition of the RSSF, it is important that these factors

never be allowed to become secondary to economic factors in the decisionmaking process. Vigorous and timely pursuit of ultimate disposal techniques would assist in negating such a possibility." (Reference 6-29, page 2)

EPA's general concern was that the AEC's concentration on the RSSF concept indicated a change in AEC's attention from the primary goal of finding an ultimate disposal method to the secondary goal of finding an interim method. EPA also felt that the technical cost/benefit was inadequate:

"The draft statement is noticeably deficient in its lack of any meaningful cost/benefit analysis supporting the RSSF option as opposed to even the other alternatives that were introduced." (Reference 6-29, page 2)

The draft environmental statement received EPA's lowest category of evaluation. Significantly one of the first actions taken by Robert Seamans after he became Administrator of ERDA was to withdraw his request for funds to build the RSSF. Like the Lyons' salt mine before it, the RSSF was officially dead. (Reference 6-30)

EPA and CEQ are already examining these arguments to see if they will still hold. Admittedly, the political climate has changed and the need for storage capacity has become a crucial factor in future nuclear decisions. However, if a SURFF and a geologic disposal facility are to be ready simultaneously, the latter is obviously preferred, especially if the costs of retrieval are similar.

6.3.3 Program Status and Plan

Following President Carter's nuclear energy policy statement on April 7, 1977, the Energy Research and Development Administration stated that:

"The policy of indefinitely postponing reprocessing leads to a requirement for storage of fuel elements in a manner in which they may be easily recovered--either for permanent disposal as waste or for reprocessing to recover fuel values after an indefinite period. An essential feature of such a storage technology is that the fuel elements should be packaged in such a manner to cause minimum complication to reprocessing consistent with safe storage. For such a purpose an engineered facility located on the surface appears at this time to be the preferred technical approach. Accordingly a Spent Unreprocessed Fuel Facility or SURFF for storage of spent fuel will be added to the ERDA program. The development of this SURFF is expected to proceed concurrently with the basic geologic repository program. The development of SURFF will be paced so as to permit the operation of such a unit by 1985. The SURFF like the repository is expected to be licensed and regulated by the NRC." (Reference 6-24, page 3, emphasis added; see also Reference 6-25, page 21).

Recent conversations show that this is a development program in flux. Carl Cooley, the project director for SURFF, says that SURFF is presently only an option being developed for consideration. He does not expect a final decision to be made on the acceptability of using a SURFF until 1979.* The "program" for doing evaluation studies is at the preliminary stages. Indeed, no budget decision on how the FY 1978 commercial waste management budget of \$175,000,000 will be allocated to SURFF studies has been made.

This statement also agrees with a careful reading of the National Energy Plan (Reference 6-26, page 72). The Plan makes no mention of an engineered surface facility but only of using geologic repositories to store spent fuel (see next section) and improving methods of storing fuel so that current storage capacity can be doubled "without constructing new facilities" (Reference 6-26, page 73).

6.3.4 Issues

Although both the NAS review of the RSSF concepts and the Technical Alternatives Document's description of the interim storage of spent fuel state that the technology is available, some outstanding issues and questions should be addressed. Because ERDA projects to examine SURFF as an option are just beginning, this is only an indication of what component issues and questions are being addressed. This is not the same as saying the issues will be resolved or the questions answered, or even whether the questions and issues are important. The four areas addressed are:

- The effects of reactor environment on the material integrity of fuel rods
- Technical issues addressed in the NAS review of the RSSF

6.3.4.1 Scientific Feasibility - The Effects of Reactor Environment on the Fuel Rods' Material Integrity. This section has frequently noted the multiple barrier concept used in the SURFF concepts. The fuel cladding which is the first barrier is exposed to extreme temperatures and radioactivity over at least a two year period in the reactor core. Does this exposure detract from the cladding's ability to confine fission products? Presently there appears to be little scientific evidence for this in the literature. The obvious, but possibly unproven assumption is that the cladding, having survived the extremes, can therefore survive lesser environments for 100 years.

This question will probably be addressed in the future studies by ERDA through their Richland Office. In addition, adequate monitoring capability should allow the facility operators to identify initial confinement breakdowns prior to any damage to the other barriers.

*Telephone conversation: C. Cooley, ERDA with E. Edelson, JPL, June 23, 1977

Although this now becomes an engineering management issue -- discussed below -- it is disturbing to note in Reference 6-21 the following description of maintenance of an air-cooled vault:

For the first several years of storage, a program of routine inspection could be adopted to assess the integrity of the container. The need for inspections would presumably decrease should experience show that container integrity was not diminishing with time. (Reference 6-21, page 17.33)

It appears that the opposite argument -- having been subjected to the extremes of a reactor core, the cladding susceptibility to deterioration over time increases -- is just as valid. This would therefore require either an equal or an increasing number of inspections as time goes by.

6.3.4.2 Technical Feasibility -- Related Technical Issues Addressed in the NAS Review of the RSSF. This section has indicated the present lack of detailed technical designs for SURFF. In addition, the similarities between SURFF and RSSF have been noted. For these two reasons it is useful to review the assessment of the RSSF made in 1975 by the Committee on Radioactive Waste Management (CRWM) of the National Academy of Sciences-National Research Council.

The CRWM called the accumulation of radioactive waste a "Challenging, but resolvable, problem...Sufficient knowledge is available to proceed immediately with developing a plan for interim, retrievable, and safe surface storage." (Reference 6-19, page vii)

However, as one of its ten conclusions and recommendations, the CRWM listed nine areas where "final specifications and designs have not been determined and considerable additional engineering research and development are necessary." Seven of these apply to SURFF. They are:

- (1) Study of performance of each of the cited barriers, including optimum materials for canister and overpack, and optimum specifications for concrete under the various environments of operation
- (2) Study of possible deleterious interactions between components of the waste and canister, overpack, or concrete -- each extrapolated to about 100 years
- (3) Procedures for fabricating or processing reliable and economical components of the demonstration facility
- (4) Study of criteria for nondestructive testing of the components of the system
- (5) Procedures for filling, sealing, overpacking, and transporting the canisters

- (6) Procedures for monitoring thermal performance during all stages of the demonstration facility
- (7) Procedures and techniques for the early detection of leaks or other failures of the canister, cask, concrete shield, or other structural members of the storage unit (Reference 6-19, page 3)

Each of these areas must be addressed in sufficient detail before the technology can be considered technically achievable.

Another technical feasibility question is why was the near-surface heat sink never recommended for high-level waste storage? Although this question might have a simple answer, at this time there is no clear indication why this alternative which "could probably be constructed and be receiving fuel sooner than would facilities based on any other concept" was never evaluated in the Reference 6-22 for RSSF. The heat generation capacity of storage containers for RSSF and SURFF is very similar. Radioactivity does not appear to differ much either. The questions raised by the omission may indicate some technical or legal problems concerning this concept that need closer examination in the SURFF design studies.

6.4 DEEP GEOLOGIC DISPOSAL OF UNREPROCESSED SPENT FUEL IN BEDDED SALT

6.4.1 Description of Repository and Packaging Facility Concepts.

In concept, a repository for the long-term containment and isolation of spent fuel is similar to the repository for solidified, encapsulated HLW described in Section 5.3. However, the volume of spent fuel per GW(e)-year is about four times the volume of the corresponding solidified HLW (including fuel rod cladding). The length and volume of a spent fuel assembly, the method of packaging spent fuel in canisters for long-term storage, and the "decay heat" and radioactivity content per canister all influence major repository design features, such as diameter of the access shafts, height of the disposal rooms or "vaults," canister spacing, etc. [See Section 6.4.4 below]. A major design parameter is the duration of pre-storage, or "aging" before emplacement in the repository. This parameter determines the level of decay heat and radioactivity of the spent fuel at the time of its emplacement in the repository. The costs of pre-storage have to be weighed against the costs and technical difficulties associated with the bedded-salt repository as a function of the duration of pre-storage.

At the present stage of development, the packaging facility conceptual design must be versatile enough to handle at least two different kinds of spent fuel shipment received:

- (1) Unpackaged spent fuel assemblies or fuel rod bundles shipped to the repository in casks from spent fuel pools at reactor sites, or from central spent fuel pools [Section 6.1 and 6.2].

- (2) Packaged spent fuel assemblies shipped from SURFFs; these SFAs will generally require repackaging [Paragraphs 6.2 and 6.3].

After discussing the brief history of geologic disposal of spent fuel and the present national program and plans in Section 6.4.2, the major design parameters for the repository and packaging are analyzed in Section 6.4.3:

6.4.2 Developmental History and Program Status and Plans

From the time of the first report in 1957 of an advisory committee of the NAS-NRC (Reference 6-33) up to April 7, 1977, the primary emphasis of the national terminal waste storage program had been on the development of a facility to bury high-level radioactive wastes from the chemical reprocessing of spent fuel elements in bedded salt deposits. Of course, it was always understood that most of the scientific and technical problems associated with such a repository would be quite similar if unprocessed spent fuel was to be contained and isolated for long times. The differences appear in packaging, in detailed canister and repository design, and in long-term radioactivity content (Section 6.4.3). However, President Carter's policy statement of April 7, 1977 deferring reprocessing for an indefinite period of time has had a major impact on the waste program, and this impact is not fully reflected in the brief discussion of this question in "the National Energy Plan". To quote from that document (Reference 6-34, page xxi):

"To ensure that adequate waste storage facilities are available by 1985, ERDA's waste management program has been expanded to include development of techniques for long-term storage of spent fuel. Also, a task force will review ERDA's waste management program. Moreover, improved methods of storing spent fuel will enable most utilities at least to double their current storage capacity without constructing new facilities."

A very brief statement on pp. 72 and 73 of this document adds some target dates:

"Prototype technologies, complete designs, and initial environmental criteria for waste repositories will be developed by 1978. Licensing of the first repository should be completed by 1981. There will be an opportunity for thorough public review at each of these stages. A task force under the direction of the Assistant to the President for energy* will review the entire ERDA waste management program."

*Dr. James P. Schlesinger (now Secretary of the Department of Energy)

An extensive public statement on May 16, 1977 by Dr. Colin Heath, of ERDA³ (Reference 6-35) provides some elaboration of these policy and schedule guidelines contained in the National Energy Plan. The major milestones in the schedule for the 1985 repositories are shown in Table 6-3, taken directly from Figure 4 of Dr. Heath's statement. As explained by Dr. Heath, in order for ERDA to be able to start construction, the agency will have to obtain construction licensing approval from NRC by 1981. "This action will be a clear signal to the American public that the program is proceeding to a complete technical solution." (pp. 11 and 12). Later on (pp. 20-21), Dr. Heath states that:

"An operating repository in salt by 1985 remains as the primary goal of the Terminal Storage Program. Hence, we do not expect any change in that part of the geological program directed at the 1985 repository as a result of the new national nuclear policy. The emphasis is being altered, however, to give first consideration to spent fuel elements as the waste form rather than solidified waste from the reprocessing plant. Thereby, a disposal option will be provided in the event that a decision is ultimately made to permanently forego reprocessing and treat spent fuel elements as waste. Capability to accept solidified waste from a reprocessing plant will also be included but primary emphasis will be on spent fuel elements."

On page 21, Dr. Heath also signals a significant shift in emphasis in the site selection program. Because of the indefinite suspension of reprocessing and the slower-than-planned build up of LWR plants, the pressure for multiple repositories is lightened. Consequently, the development of the four additional sites in crystalline rock or argillaceous formations (shale) may be "stretched out" in time, although basic work on the geology of these alternatives would continue. Clearly the primary emphasis is on the development of two Federal repositories in bedded salt by 1985, largely because of time constraints and the more highly developed state of technology for bedded salt as compared to other alternatives. The emphasis on spent fuel elements as a "residual", either for an interim period or for all time, is reflected in a considerable increase by ERDA in the efforts to develop packaging methods for handling and storing spent fuel.

An important element of the waste management program is the preparation by ERDA of the Generic Environmental Impact Statement (GEIS), evaluating the potential releases to the environment of commercial post-fission radioactive materials--either spent fuel or high-level wastes from reprocessing and recycling operations with uranium only, or with both uranium and plutonium. The activities examined are the major ones of storage, treatment (if any), transportation, and final disposition. A draft of this GEIS is expected to be issued in late 1977. Following public hearings, a final version of the GEIS will be prepared in 1978, and this document will serve as a guide to future decisions on the general nature of the waste repository program. Of course, the GEIS does not eliminate the need for a site-specific EIS.

Table 6-3. Terminal Storage Program

Goals		
●	Initiate major industry participation	1977
●	Identify suitable geologic formations	1978
●	Waste acceptance criteria and charges	1978
●	Site selection -- first repositories	1978
●	Draft EIS and firm facility cost estimate	1979
●	Start construction -- first repositories	1981
●	First repositories operational	1985

6.4.3 Technical Aspects of Spent Fuel Repository Design and Packaging; Major Design Parameters

6.4.3.1 Radioactivity and Decay Heat of Spent Fuel. With a "burn-up" rate of 33,000 MW(t) days per tonne, an energy density of 30 MW/tonne and an average load factor of 80% the total radioactivity content of the transuranics in the 27 tonnes of spent fuel is 3×10^6 Ci one year after discharge, compared to a value of 10^8 Ci for the fission products. (Reference 6-36) Most of this radioactivity in the transuranics comes from β -radiation emitted by ^{241}Pu which has a half-life of 13.2 years. After ten years, the total radioactivity content in the transuranics has declined slightly to 2×10^6 Ci, while the radioactivity in the fission products has decayed to 10^7 Ci.

During the next 90 years, ^{241}Pu virtually disappears. One of the main contributors to β -radiation, ^{244}Cm (half-life 17.6 years), decays by a factor of 50, and ^{238}Pu (half-life 86 years) decreases by a factor of about 2.23. The net result is that the total radioactivity content of the transuranics in the spent fuel after 100 years is 1.3×10^5 Ci, compared to a value of about 10^6 Ci for the fission products.

During the first 100 years of storage, the total radioactivity of the spent fuel is generated mainly by the fission products, and differs only slightly (15-20% higher) from the total radioactivity in the high-level waste after reprocessing. A similar conclusion naturally follows for the "decay heat". Thus, for the first 100 years, the containment problem for the spent fuel in bedded salt is not much different in principle from the problem for encapsulated HLW. The important design inputs are 9.2 kw per metric ton of spent fuel after one year of storage; 2.3 kw per metric ton after 5 years; 1.4 kw per metric ton after ten years; 0.3 kw per metric ton after 100 years (Figure 6-18).

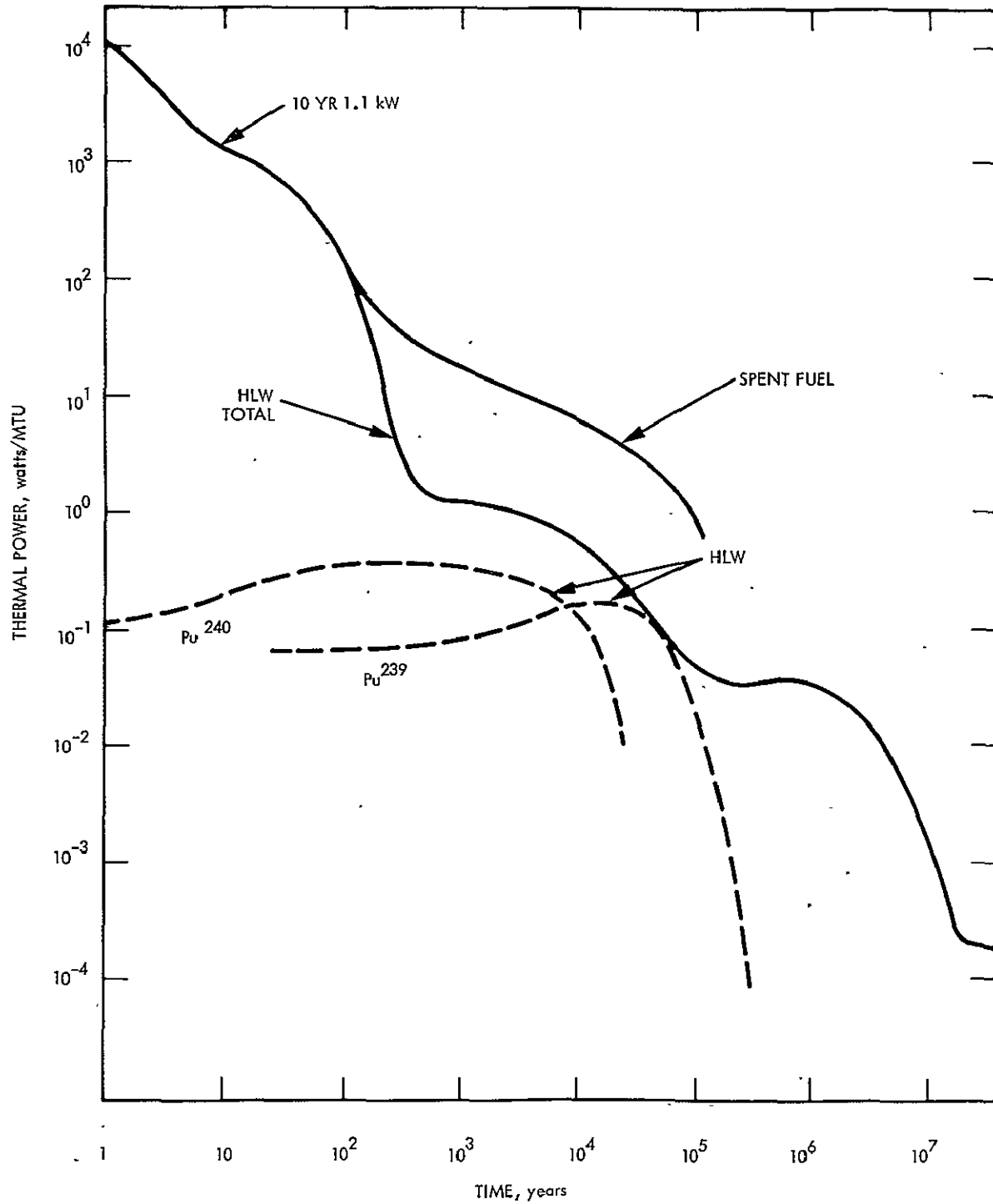


Figure 6-18. Thermal Power Per Metric Ton of Uranium Versus Time

For storage times longer than 100 years, the time histories of radioactivity content and decay heat for spent fuel and HLW differ significantly. Since the spent fuel contains 200 times as much plutonium initially as the HLW, the contributions made by ^{239}Pu and ^{240}Pu shown in Figure 6-19 in the early period, i.e., around one year after discharge from the reactor, are multiplied by a factor of 200, until ^{240}Pu (half-life 6760 years) begins to decay. The build-up of ^{241}Am in the spent fuel during the first 10 years after discharge (Reference 6-36) is also an important factor, until this isotope (half-life 458 years) also decays. (^{243}Am plays a secondary role.) The net result is that the decay heat for spent fuel is more than 10 times higher than the decay heat for HLW per metric ton of uranium in the discharged fuel over the period 10^3 - 10^5 years (Figure 6-18). Thus the integrated heat load for spent fuel over the first 10^4 years is much larger than for HLW if the areal density in the disposal layer in kw/acre (or watts/m²) is the same at the time of emplacement in the repository. Thus, the thermal design problems for the repository are broken down quite naturally into two requirements: (1) the "near-field" problem concerned with temperatures in the disposal layer itself and especially at the canister wall during the first 100 years of storage; (2) the "far-field" problem dealing with the temperature rises and thermal stresses in the entire repository and surrounding geologic medium in the period 10^2 - 10^4 years after storage.

6.4.3.2 Packaging, Canister Spacing and Temperature Effects on Bedded Salt.

6.4.3.2.1 Near-Field Considerations. The major design parameters for a spent fuel repository are as follows:

- (1) "Pre-storage" time elapsed between removal of spent fuel from the reactor and emplacement of the spent fuel in the repository.
- (2) Maximum allowable canister wall temperature rise in bedded salt (ΔT_{MAX}).
- (3) Lateral dimensions and shape of canister cross-section.*
- (4) Rate and cumulative amount of spent fuel to be shipped to repository.

These four parameters determine the amount of spent fuel and the "decay heat" per canister, the spacing between canisters, and the number and cost of the canisters; the diameter and cost of the canister access shaft (or shafts) to the repository; ease of handling, transporter design, and size and height of storage rooms or "vaults".

*Based on the assumption that spent fuel rod integrity is maintained, the minimum canister length is fixed at about 4 m.

About 65 spent fuel assemblies are discharged from a typical PWR per GW(e)-year [70% load factor], or about 2.3 assemblies per metric ton of spent fuel. Each assembly is 22 cm by 22 cm in cross-section and about 4 m long, and contains 264 fuel rods. One year after discharge from the reactor each assembly generates about 4 kw of decay heat; after 5 years this heat rate drops to 1.0 kw per assembly, and after 10 years it amounts to 0.6 kw per assembly. In dealing with the packaging problem for spent fuel, there are three main alternatives:

- (1) Single-assembly canisters
- (2) Dismantling of fuel assemblies and subsequent packaging of fuel rods/bundles in canisters to reduce volume required by about 40% compared with (1)
- (3) Multiple-assembly canisters

These alternatives must be carefully analyzed in terms of the major design parameters listed above in order to develop optimum solutions. Some of the main features of these design alternatives will be briefly described here.

Maximum temperature rise at the canister wall (ΔT_{MAX}) has been calculated as a function of initial "decay heat" (or pre-storage time), canister spacing and canister diameter for HLW canisters embedded in rock or in salt. (References 6-37 and 6-38) The field of canisters is regarded as a square, planar infinite array of cylinders embedded in a continuous, static rock or salt bed. If it were not for the fact that the heat conductivity of rock or salt depends on the temperature, the problem would be linear. However, over the modest range of temperatures of interest to the designer this effect is small and the problem is quasi-linear. Therefore one can scale the results obtained for the HLW canisters to fit the case of canisters containing spent fuel. Examining the wall temperature history for any one particular canister, one finds that the temperature will increase at first because of the heat flux from this canister and the adjacent ones. However, the effect of the more distant canisters on the wall temperature takes some time to be felt; in fact, this time interval is roughly proportional to the square of the distance and inversely proportional to the thermometric conductivity.* If the spacing between canisters is large enough, this time interval is long enough so that the "decay heat" emitted from each canister has dropped off appreciably from its initial value at the time of emplacement. Clearly, the temperature rise at the canister wall reaches a maximum value and then decreases with time thereafter. The numerical calculations show that this ΔT_{MAX} drops off quite rapidly with increasing canister spacing up to a spacing of about 10 meters (Figure 5, Reference 37). Beyond this value only slight reductions in ΔT_{MAX} are obtained at increasing cost. The maximum temperature

*Thermometric conductivity, γ is $k/\rho C_p$, where k is thermal conductivity in watts/m/°C, ρ is density in kg/m³ and C_p is specific heat in joules/kg/°C.

at the canister wall is also sensitive to the "effective diameter" of the canisters (Figure 5, Reference 37). These results can be utilized to describe some of the main features of packaging alternatives (1) and (3) above.

If the single assembly alternative is selected, 65 canisters would be required per Gw(e)-year, as compared with about 10 canisters for encapsulated HLW. These canisters could be of square cross-section (28 cm x 28 cm) and about 4 m long, filled with helium gas or zinc powder for good thermal conduction and protection of the fuel cladding against corrosion and oxidation. A stainless steel wall thickness of 1.3 cm is adequate for radiation shielding, so that each canister would have an internal volume of about 0.2 m³ and would weigh about 0.7 metric tons. If the spent fuel is packaged and placed in the repository 5 years after removal from the reactor, each canister would generate 1 kw of decay heat; if the pre-storage time is 10 years, the decay heat per canister is 0.6 kw. With a canister spacing of 10 meters, the ΔT_{MAX} at the canister wall is about 60°C for 5 years of pre-storage and about 35°C for 10 years of pre-storage (Table 6-4). These maximum temperature rises are well below the values at which any significant difficulties are expected because of brine migration in the salt, or salt decrepitation (Section 5.3.4).

On the other hand, the multiple fuel assembly mode might involve packaging 4 spent fuel assemblies, for example, in a canister of square cross-section 50 cm x 50 cm, and about 4 m long, with an interior volume of about 0.3 m³. For a wall thickness of 1.3 cm, the canister weight is about 3.2 metric tons. Only 17 such canisters are required per 27 MTU (one annual discharge) compared with 65 canisters for the single-assembly mode, and 10 canisters for encapsulated HLW. In this case, taking into account the larger "effective radius" of the canister, ΔT_{MAX} at the canister wall with a canister spacing of 10 meters is about 175°C if the pre-storage time is 5 years. This value decreases to about 105°C if the pre-storage time is 10 years (Table 6-4). Of course the diameter of the canister access shaft(s) to the repository would have to be at least 70 cm, compared with about 40 cm for the single assembly case. As we shall see shortly, this alternative of a multiple fuel assembly mode is probably not acceptable when the integrated heat loads are considered in the "far-field" around the disposal layer.

For all three packaging alternatives, the length of the canister is about 4 m if the integrity of the fuel rods is to be preserved. This length should be compared with the proposed length of about 3 m for the canisters containing encapsulated HLW (Section 5.2). The design height of a storage room or vault in the repository would have to be increased somewhat in order to accommodate canisters containing spent fuel.

6.4.3.2.2 "Far-Field" Considerations. Lincoln and Parry (Reference 6-38) have pointed out that either "near-field" thermal response in and near the disposal layer, or "far-field" long-term thermal response of the bedded salt and surrounding geologic medium may determine the permissible level of

Table 6-4. Design Characteristics for Spent Fuel Disposal

Canister Packaging Mode	Canister Cross-section	Canister Weight (MT)	No. of Canisters per Annual Discharge	T _{MAX} at Canister Wall	
				t ₀ = 5 years	t ₀ = 10 years
Single Fuel Assembly	28 cm x 28 cm	0.8	65	60°C	35°C
Multiple Fuel Assembly (4)	50 cm x 50 cm	3.2	17	175°C	105°C

Typical 1000 MW(e) PWR operating at 70% load factor:

33,000 MW(t) days/MT and 30 MW/MT

Canister spacing = 10 meters

t₀ = Time interval between discharge from reactor and emplacement in repository

"power density" in the disposal layer. From Figure 6-18 showing the time history of the decay heat, one can roughly estimate the integrated heat load for HLW and for unprocessed spent fuel over the period $10-10^4$ years, as follows: (Here ΔQ is measured in watt-years per MTU)

<u>Period (years)</u>	<u>ΔQ (HLW)</u>	<u>ΔQ (Spent. Fuel)</u>
10 - 100	25,000	30,000
100 - 1000	22,000	66,000
$10^3 - 10^4$	13,000	200,000
10 - 10^4	Cumulative $\Delta Q = 60,000$	Cumulative $\Delta Q = 300,000$

So if there were a requirement on the "far field" to maintain the integrated heat load for spent fuel and for HLW at the same levels over the first 10^4 years, the areal density in the disposal layer for spent fuel would have to be about 1/5 that for HLW.

Now, if the HLW canisters are spaced 10 meters apart on centers, there is one canister per 100 m^2 , and if 15 are utilized to contain the encapsulated HLW from 30 MTU of reprocessed spent fuel (1 GW(e)-year output), each canister emits a decay heat rate of 2.2 kw if it is emplaced in the repository ten years after discharge from the reactor. In other words, the power density is 22 w/m^3 , or 110 kw/acre. In order to produce the same integrated heat load for spent fuel over 10 years, the power density would have to be about 4 w/m^2 or 22 kw/acre. It is interesting to note that this power density is matched very closely by adopting the single fuel assembly per canister packaging mode discussed in Section 6.4.4.2.1. Of course the number of canisters required per GW(e)-year is increased from 15 for HLW to 75 for spent fuel. It should be emphasized that with these integrated heat loads, the present 1-D models show temperature rises in the range of only $10^\circ - 15^\circ \text{C}$ at a distance of 500 m above the disposal layer after 10^4 years of storage.

All of the thermal calculations known to the present authors utilize a one-dimensional model. This approach is certainly justified for the first 100 years of storage, because the "heat pulse" from the disposal layer will have produced appreciable temperature increases out to distances of about 100 m away from the disposal layer, which is assumed to be a planar array of the order of $3000 \text{ m} \times 3000 \text{ m}$. But after 10^4 years, the "heat pulse" will be felt out to a distance of the order of 1000 m, and a 3-D analysis is required. A schematic diagram of the isotherms around the disposal layer is shown in Figure 6-19 for storage times of the order of 1000 years. Clearly there are steeper thermal gradients around the edges of the disposal layer than in a vertical mid-plane. These thermal gradients, acting over long periods of time, would be accompanied by stresses which could produce undesirable cracks or faulting in the bedded salt or surrounding geologic medium. One method of minimizing this possibility is to "taper" the distribution of power density in the disposal layer toward the edges. The phenomena

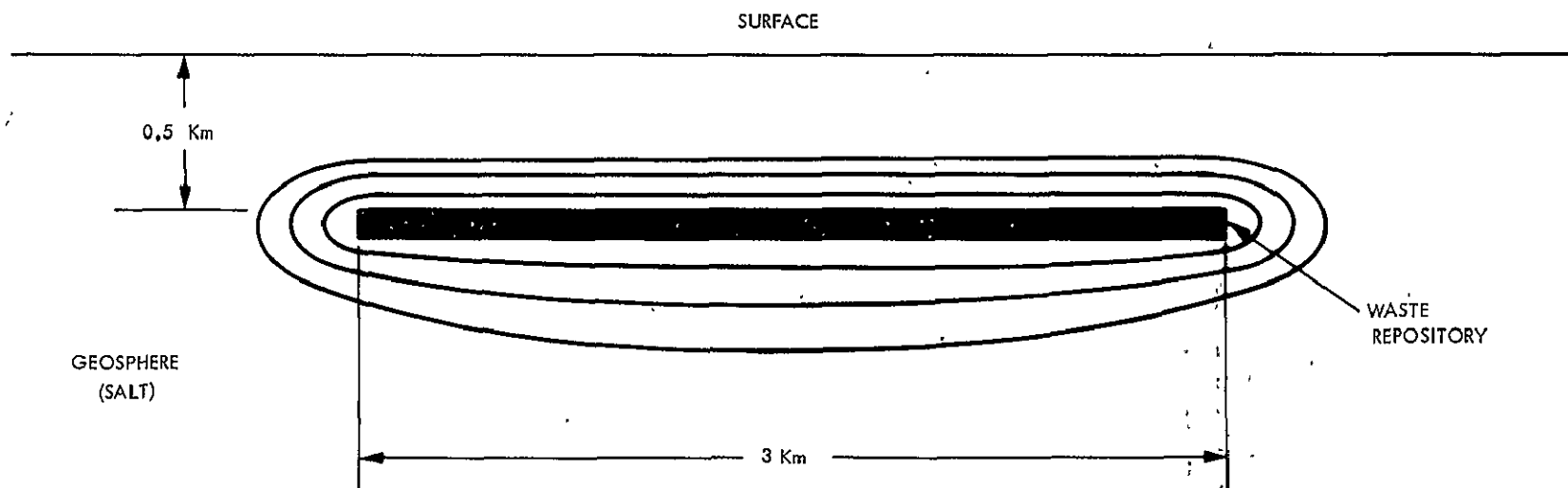


Figure 6-19. Schematic of Isotherms for HLW Disposal Site

associated with the long-term response of bedded salt and surrounding earth and/or rock to modest temperature gradients requires careful study utilizing a more realistic 3-D model.

For the "open" LWR nuclear fuel cycle, the total number of canisters and the size of the repository (or repositories) required for containment and isolation of spent fuel over the entire life of the LWR can be estimated on the basis of these design considerations. Recent estimates of total U.S. reserves of U_3O_8 made by a special panel of the National Academy of Sciences places these reserves (known, probable, possible and suspected) at about 2×10^6 tons. Using current technology this reserve would produce 3×10^5 tons of UO_2 fuel enriched to a ^{235}U content of 3.5%. If the multiple-assembly packaging alternative were adopted (for example), about 1.7×10^5 canisters would be required. These canisters could be arranged in a square array (plan view) with a spacing of 10 meters between canisters. The volume occupied by the canisters alone is $1.4 \times 10^5 m^3$. The entire array would occupy an area 4.2×10^3 meters by 4.2×10^3 meters, or 17 square kilometers (about 6 square miles). According to the revised National Waste Terminal Storage (NWTS) program, the spent fuel would most likely be stored in at least two separate repositories in bedded salt. In that case, each storage component of a repository would occupy an area of about $3 mi^2$, or about 2000 acres. If the "far-field" thermal considerations discussed above lead to the alternative mode of packaging a single fuel assembly in one canister then obviously 4 times as many canisters are involved and 8 separate repositories each occupying a disposal layer area of 2000 acres are required.

6.4.3.3 Long-Range Considerations

Clearly, the key questions about the viability of geologic containment and isolation of radionuclides are concerned with the engineering design of the repository, based on the hydrology and geology over long periods of time, and the possible pathways to the biosphere.

In Section 5.3.4 the weak links in the repository site evaluation and selection process are outlined, and several possible scenarios are constructed in which a connection between the repository and the biosphere could be established. The most important geospheric transport mechanism is the intrusion of flowing ground water. If the hydrogeological criteria for site stability can be established, and the existing ground water systems measured and modeled for a specific site (see Section 5.3), the probability of the intrusion of ground water in a carefully selected and designed bedded salt repository over the first 10^3 years is virtually zero. Many examples can be found in human history of the successful preservation of entombed objects for several thousand years. A containment period of 10^3 years is adequate to insure that practically all the fission products of any consequence in the HLW or the spent fuel have disappeared with the exception of ^{129}I and Technetium 99.

The probability of ground water intrusion over time intervals longer than 10^3 years is low; but it is difficult to estimate with any degree of confidence. Therefore it is useful to make the conservative

assumption that such ground water flow does in fact take place, and then to examine the main geologic parameters that control the rate of transport of the long-lived radionuclides to the nearest interface with the biosphere. The most important index of the effectiveness of the geologic barrier for any particular radionuclide is the ratio of the geospheric transport time to the half-life of the nuclide ($\tau_{1/2}$). This transport time depends upon two parameters:

- (1) The "quality" of the geologic formation in terms of its permeability, P in m/sec, and porosity, or void fraction, ϵ . These parameters are combined to give the mean pore velocity of the mobile phase, $V_p = Ph^1/\epsilon$ (in meters/sec), where h^1 is the hydraulic gradient in meters of "head" per meter of path length;
- (2) The "hold-up" factor, K , or the ratio of mean pore velocity to the nuclide migration velocity. This factor depends on the degree of sorption of the radionuclide in the geologic barrier by the processes of ion exchange, ion absorption, etc.*

The ratio (R) of the geospheric transport time to the half-life of the nuclide is then

$$R = 3 \times 10^{-8} \frac{\left(\frac{L}{V_p}\right) K}{\tau_{1/2}}$$

where L is the path length to the nearest interface with the biosphere in meters, and $\tau_{1/2}$ is measured in years, e.g., for ^{239}Pu , $\tau_{1/2} = 25,000$ years. Neglecting diffusion and dispersion the "exit" radioactivity concentration in the flowing groundwater is related to the initial concentration at the repository by the expression $(\text{Ci/m}^3)_{\text{EXIT}} = (\text{Ci/m}^3)_0 e^{-0.69R}$.

The present brief discussion is not concerned with the ultimate fate of the radioactivity (in Ci/m^3) along its pathway to man, but we will utilize the "exit" value of (Ci/m^3) in comparison with the maximum permissible concentration (MPC) of a particular radionuclide as an index of geospheric containment. Following Marsily, et al (Reference 6-54) the simplest and most probable situation involves

*The hold-up factor K can be shown to be given by the relation $K = 1 + K_d \rho / \epsilon$, where ρ is the mass per unit volume of the immobile phase, ϵ is the effective porosity, and K_d , the "distribution coefficient," is the ratio of the concentration of the particular element per unit mass in the immobile phase to the concentration of this element per unit volume of the mobile phase, under equilibrium conditions (see References 6-52 and 6-53).

upward flow of groundwater across the repository disposal layer in response to a vertical hydraulic gradient. In that case the radioactivity concentration of a particular nuclide in the groundwater at the repository is given by

$$(Ci/m^3)_0 = 3 \times 10^{-8} \frac{M_N(Ci/g)_N}{T_{L-D}(\epsilon V_p)(m^2/CANISTER)}$$

where M_N is the mass of a specific nuclide per canister in grams, $(Ci/g)_N$ is the curies per gram of a specific nuclide, T_{L-D} is the leaching or dissolution time for all of this nuclide to be discharged into the water in years, ϵV_p is the Darcy velocity in m/sec, or flux of water/m² across the disposal layer. For example, for spent fuel with an area distribution of 1 canister per 100 m², and a relatively poor geologic formation characterized by $\epsilon = 0.02$ and $V_p = 5 \times 10^{-6}$ m/sec, the concentration for ²³⁹Pu at the disposal layer is

$$(Ci/m^3)_0 \cong \frac{0.4}{T_{L-D}},$$

so that

$$(Ci/m^3)_0 = 4 \times 10^{-3} \text{ if } T_{L-D} = 100 \text{ years, and}$$

$$(Ci/m^3)_0 = 4 \times 10^{-2} \text{ if } T_{L-D} = 10 \text{ years.}$$

Thus attenuation factors of the order of 10^{-3} to 10^{-4} are required along the transport path to the environment in order to maintain the exit level of radioactivity concentration below the MPC. In other words the "hold-up" factor K must be large enough so that the transport time for ²³⁹Pu is 10 to 14 half-lives for this particular formation, or 250,000 to 350,000 years.

Although this simplistic discussion turns out to contain the essential features of the problem, the actual details of the situation including diffusion and dispersion and the nuclear chain reactions in the spent fuel or HLW are quite complicated. Radionuclide migration in geologic media has been analyzed by Schneider and Platt (Reference 6-38) and by Burkholder in the U.S. (Reference 6-53), and by Marsily, Ledoux, Barbreau and Margat in France (Reference 6-54) using one dimensional models.

Burkholder utilizes fixed values of K and V_p . In his work $V_p = 1$ foot per day $\cong 4 \times 10^{-6}$ m/sec and (for example) K for ²³⁹Pu is taken as 10^4 for a non-salt repository, corresponding to the best available value of K_d (see Footnote, pg. 6-60) for U.S. western desert soil. Burkholder's calculations illustrate the combined effects of path length, canister integrity (or time to failure), and leach rate (for HLW) or discharge rate at the repository, on the leakage of radioactivity in terms of Ci/year. Burkholder is primarily concerned with the possibility of horizontal transport of radionuclides by the

groundwater to streams or rivers that are located 10-100 km away from the repository. The nuclides released from the repository are assumed to be diluted by a river flowing at a rate of 10^4 cubic feet per second (about 7×10^6 acre-feet/year). These nuclides are assumed to accumulate in this medium for 50 years at their peak discharge rates, and the individual receiving the maximum dose is assumed to be exposed to that 50-year accumulation for another 50 years. In view of the simplified discussion given above, it is not surprising that with his assumed values of K , reasonable combinations of canister integrity, leach rate and path length can be found for which the expected incremental "dosage" to the "maximum individual" is no larger than the "background" level taken to be 120 in rem/year. However, as Burkholder himself points out, the values of the hold-up factor K under actual geological conditions for a bedded salt repository are not known at present.

The French paper by Marsily, et al, adopts a quite different point of view--the authors are concerned mainly with the relative importance of the "quality" of the geologic formation, as expressed by V_p and ϵ , and the hold-up factor K . They regard vertical movement of ground water as the most likely transport mechanism, because significant vertical gradients of piezometric head are often found. The distance from the repository to the surface is taken as 500 meters. They consider five possible geologic formations listed in order of increasing quality in Table 6-5. In order to study the extreme cases they take two values of K for ^{239}Pu ; $K_d = 0$ and $K = 1$ (no sorption), and $K_d = 2000$ ml/g, or $K = (5 \times 10^3)/\epsilon$ (strong sorption). For ^{237}Np they take $K_d = 15$ ml/g, or $K \cong 40/\epsilon$, while for ^{129}I it is well-known that $K_d \cong 0$ and $K = 1$. The results including dispersion-diffusion labeled "Hypothesis 2" in Tables 6-6 and 6-7 tell the story. Under Hypothesis 2 the glass encapsulating the HLW retains its integrity for 10^4 years, and then the glass matrix structure is damaged and the total load of HLW is released into the leaching water over the next 5000 years (T_{L-p}). In these calculations the areal density is 1 canister per 25 m^2 and $M_N \cong 100 \text{ g}$ of ^{239}Pu per canister.

The following significant conclusions can be drawn from the results of this simplified calculation:

- (1) Without sorption, even an excellent confining formation (Case 4) only introduces a delay in the transport of the radionuclide to the environment; it does not retain it. [In Case 1 the transport time to the surface for an appreciable concentration without sorption is about 5 years; in Case 4 it is about 14,000 years.]
- (2) With a high degree of sorption ($K_d = 2000$) even a relatively poor geologic formation (Case 1) retains the ^{239}Pu for a sufficiently long time so that radioactive decay prevents it from reaching the environment.

These conclusions would be qualitatively the same for spent fuel, because the attenuation factor for $K_d = 2000$ is of the order of 10^{-6} for Case 1, and very much smaller for Cases 2-4. In fact, one could go one step further, and point out that there is a direct proportion-

Table 6-5. Parameters of the Geologic Formations

Geologic Formation	Darcy's Permeability (m/sec)	Hydraulic Gradient	Effective Porosity (%)	Resulting Velocity of Water	
				Darcy's (m/sec)	Mean Pore (m/sec)
1	10^{-6}	1/10	2	10^{-7}	5×10^{-6}
2	10^{-6}	1/50	2	2×10^{-8}	10^{-6}
3	10^{-7}	1/50	5	2×10^{-9}	4×10^{-8}
4	10^{-8}	1/50	10	2×10^{-10}	2×10^{-9}
5	10^{-10}	1/50	20	2×10^{-12}	10^{-11}

Table 6-6. Plutonium Migration under Hypothesis 2 (Not Sorbed)

Geologic Formation	Geologic Formation (Velocity of Water m/sec)	Ratio of Concentration to its MPC	Years until Maximum is Observed
1	5×10^{-6}	1.3	10,000
2	10^{-6}	6.0	10,000
3	4×10^{-8}	66.0	10,700
4	2×10^{-9}	470.0	20,000
5	10^{-11}	8.5×10^{-10}	700,000

Table 6-7. Plutonium -239 Sorbed

Geologic Formation	Ratio of Concentration to its MPC	Years until Maximum is Observed
1	3×10^{-9}	460,000
2	1.4×10^{-24}	1,150,000
3	0	
4	0	
5	0	

ality between the hold-up factor, K , required and the mean pore velocity. For example, for Case 3 ($V_p = 4 \times 10^{-8}$ m/sec and $\epsilon = 0.05$) a value of $K = 700$ or $K_d = 15$ ml/g would be sufficient to duplicate the attenuation achieved in Case 1 with $K_d = 2000$ ml/g.

The degree of sorption depends critically on the complex geochemistry of plutonium. When plutonium is in its usual valence states of +4 or +6, it will be strongly sorbed. But the concentrations of plutonium in the flowing groundwater are very low in the cases just considered, and the presence of small amounts of negative ions (silica or carbonate, for example) may lead to the formation of complex plutonium molecules that are electrically neutral and therefore not sorbed. These observations show the necessity for a comprehensive program of laboratory and field studies on the interactions of radionuclides with both individual minerals and rocks. Because of the extreme sensitivity of the exit level of the radionuclide in Ci/m³ to mean pore velocity, hold-up factor and path length when R is of the order of 10-20, credible predictions of radionuclide behavior cannot be made without this information.

At the present time the Scientific Feasibility of deep geologic disposal of both encapsulated high-level waste and unprocessed spent fuel in bedded salt has not yet been proven. However, there is no scientific reason to anticipate that existing, planned and recommended work will not create a scientific data base that will reduce the hydro-geological uncertainties to acceptably low levels, although the proposed time schedule may be too optimistic.

REFERENCES

- 6-1. Final Safety analysis Report, San Onofre Nuclear Generating Station, Units 2 & 3, Volume 13, Southern California Edison Company and San Diego Gas & Electric Company.
- 6-2. Final Safety Analysis Report, San Onofre Nuclear Generating Station, Unit 1, Southern California Edison Company and San Diego Gas & Electric Company.
- 6-3. "Analysis of Potential Accidents Not Involving the Core," Appendix I, Nuclear Regulatory Commission, October, 1975.
- 6-4. Code of Federal Regulations, 10 CFR 71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions", U. S. Government Printing Office, Washington, D. C., 1972.
- 6-5. Carter, T. J., "Spent Fuel Storage", Ninth Annual National Conference on Radiation Control, Seattle, WA, June, 1977.
- 6-6. Rubenstein, H. J., Clark, P. M., Gikrest, J. D., "Spent Fuel Storage Basin for New Nuclear Power Plants", American Nuclear Society Transactions, Vol 23, June, 1977.
- 6-7. Independent Spent Fuel Storage Facility SWECO-7601, Stone & Webster Engineering Corporation, Boston, Mass., June, 1976.
- 6-8. Guidance on the License Application, Siting Design, and Plant Protection for an Independent Spent Fuel Storage Installation, Regulatory Guide 3.24, Atomic Energy Commission, December 1974.
- 6-9. Woodhall, C. B., Hook, N. B., Powell, F. P., "Spent Fuel Storage in the USA," American Nuclear Society Transactions, Vol 25, June, 1977.
- 6-10. Allison, M. M., Denver, D. J., Slifer, B. C., "A High-Density Spent Fuel Storage Rack Design for Boiling Water Reactors", American Nuclear Society Transactions, Vol 23, June, 1976.
- 6-11. Grimm, E. S., Zahn, L. L., "High-Density Fuel Storage for Boiling Water Reactors", American Nuclear Society Transactions, Vol 25, June, 1977.
- 6-12. Olszewski, J. S., Clark, R. H., "BCW-Designed Closed-Spaced Fuel Storage Racks", American Nuclear Society Transactions, Vol 25, June, 1977.
- 6-13. Spier, E. M., Kamiuski, D. J., Rawley, C. J., Nowicki, J. A., "Westinghouse High-Density Spent Fuel Rack Design", American Nuclear Society Transactions, Vol 25, June, 1977.

- 6-14 Jones, T. A., Gauslow, R. W., Bevilacqua, F., "HI-CAP High-Capacity Fuel Rack Design", American Nuclear Society Transactions, Vol 25, June, 1977.
- 6-15 Consolidated Safety Analysis Report for Morris Operation, NEDO-21326-1, General Electric Company, Morris, Ill., January, 1977.
- 6-16 Summary Environmental Report, Fuel Storage Facility Expansion for Morris Operation, NEDO 21624, General Electric Company, Morris, Ill., April, 1977.
- 6-17 Independent Spent Fuel Storage Facility SWECO-7601, Stone & Webster Engineering Corporation, Boston, Mass., June, 1977.
- 6-18 Environmental Impact Appraisal By the Division of Fuel Cycle and Material Safety Related to License Amendment For Materials License No. SNM-1265 Morris Operation Facility Grundy County, Illinois For General Electric Company, Docket No. 70-1308, U. S. Nuclear Regulatory Commission, December, 1975.
- 6-19 Panel on Engineered Storage, Committee on Radioactive Waste Management, National Research Council, National Academy of Sciences, Washington, D.C.
- 6-20 Atlantic Richfield Hanford Company and Kaiser Engineers, Retrievable Surface Storage Facility Alternative Concepts, Engineering Studies, ARH-2888 REV, July 1974, p.7-2.
- 6-21 Alternatives for Managing Wastes from Reactors and Post Fission Operations in the LWR Fuel Cycle, ERDA-76-43, May 1976.
- 6-22 Environmental Statement, Management of Commercial High Level and Transuranium-Contaminated Radioactive Waste, WASH-1539 (Draft), USAEC, September 1974, p. 5.2-1.
- 6-23 Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle NUREG-0116 (Supp. 1 to WASH 1248), October 1976, Section 4.2.5.
- 6-24 Kuhlman, Carl W. "ERDA Waste Management Program" Address to the Atomic Industrial Forum Fuel Cycle Conference - Kansas City, April 26, 1977.
- 6-25 Heath, Colin A. Statement before the Wisconsin Assembly Environmental Protection Committee's Hearings, May 16, 1977.
- 6-26 National Energy Plan, Executive Office of the President, May 1977.

- 6-27 Telephone conversation, Harry Soule, ERDA and Ed Edelson, JPL, June 6, 1977. Mr. Garanson, ERDA Richland Operations Office, stated in a July 11 telephone conversation that only the SSCC and the Near Surface Storage are presently being pursued.
- 6-28 "History and Interpretation of Radioactive Waste Management in the United States", Hilberry, N., Appendix D in NRC's Goals for Waste Management.
- 6-29 EPA response to Draft WASH-1539, Nov. 15, 1974, (unpublished).
- 6-30 Letter from R.C. Seamans, Jr., to Honorable John O. Pastore; April 9, 1975.
- 6-31 Nelson, D.C. and Wodrich, D.D. "Retrievable Surface Storage Facility in Commercial High-Level Waste", Nuclear Technology, Volume 27, December 1974, p. 391.
- 6-33 "Disposal of Radioactive Wastes on Land," Publ. 519, National Academy of Sciences--National Research Council, Washington, D.C., 1957.
- 6-34 "The National Energy Plan," Executive Office of the President, Energy Policy and Planning, Wash., D.C., April 29, 1977.
- 6-35 Dr. Heath is Assistant to the Deputy Director, Civilian Waste Management, Reprocessing and Recycle; Division of Waste Management, Production and U.S. ERDA. This statement was presented at the Wisconsin Assembly Environmental Protection Committee's Hearings on the Transportation and Disposal of Radioactive Materials, Madison, Wisconsin.
- 6-36 Pigford, Thomas H., and Ang, K. P., "The Plutonium Fuel Cycles," Health Physics, Volume 29, October 1975, pp. 451-468 (Pergamon Press).
- 6-37 Cohen, Bernard L., "High Level Radioactive Waste from Light-Water Reactors," Rev. Mod. Phys., 49, pp. 1-20 (1977). (See also "The Disposal of Radioactive Wastes from Fission Reactors," Scientific American, Volume 236, No. 6, June 1977, pp. 21-31.).
- 6-38 Schneider, K. J., and Platt, A. M., "High Level Radioactive Waste Management Alternatives," Volume 2, Battelle Pacific Northwest Laboratories Report, BNWL-1900 (May 1974), Richland, Washington, 99352.
- 6-39. Johnson, A. B. Jr, "Behavior of Spent Fuel in Water Pool Storage," BNWL-2256, Battelle Pacific Northwest Laboratories, Richland, WA, 99352, September, 1977.
- 6-40. "Industry, Congress Call For White House Action on Spent Fuel Storage", Nuclear Fuel, 2, 16, August 8, 1977.

- 6-41. Correspondence with David Dawson, Nuclear Energy Division, General Electric Company, San Jose, CA, 95125, June 23, 1977.
- 6-42. Correspondence with David Dawson, Nuclear Energy Division, General Electric Company, San Jose, CA, 95125, July, 1977.
- 6-43. "Report of Preliminary Site Exploration Proposed Facility, Grundy County, Illinois for the Flour Corporation Ltd", Dames & Moore, Chicago, Illinois 60606, September, 1966.
- 6-44. "Report of Foundation Investigation, Proposed Reprocessing Plant Project, Near Morris, Grundy County, Illinois for the General Electric Company", Dames & Moore, Chicago, Illinois 60606, December, 1967.
- 6-45. Earthquake and Wind Analysis - Plant Effluent Stack, H. J. Sexton & Associates, Engineers, San Francisco, California, November, 1968, contained in Consolidated Safety Analysis Report for Morris Operation NEDO-21326-1, General Electric Company, Morris, Ill., January, 1977.
- 6-46. Earthquake and Wind Analysis - Underground Structures, H. J. Sexton & Associates, Engineers, San Francisco, California, March 1969, contained in Consolidated Safety Analysis Report for Morris Operation NEDO-21326-1, General Electric Company, Morris, Ill., January, 1977.
- 6-47. "Safety Basis for the MFRP Fuel Bundle Storage Baskets", Battelle Pacific Northwest Laboratories, Richland, Washington 99352, May 1975.
- 6-48. "Fuel Storage System Design Report GE Morris Operation", Programmed & Remote Systems Corporation, St. Paul, Minnesota 55112, April - May, 1975.
- 6-49. Seismic Shock Environment Test of Simulated Nuclear Fuel Storage Basket, Department of the Army, Construction Engineering Research Laboratory, Champaign, Illinois, August, 1975, contained in Consolidated Safety Analysis Report for Morris Operation, NEDO-21326-1, General Electric Company, Morris, Ill., January, 1977.
- 6-50. Fuel Recovery Operation Quality Assurance Plan, General Electric Company, San Jose, California 95125, May, 1976, contained in Consolidated Safety Analysis Report for Morris Operation, NEDO-21326-1, General Electric Company, Morris, Ill., January, 1977.
- 6-51. NEDO-20969, "Operating Experience Irradiated Fuel Storage GE Morris Operation," K. J. Eger, A. Hulsey General Co., August 1975. Docket Number 70-1308.

- 6-52 Lincoln, R.C. (Sandia) and Parry, S.G.S. (NRC), "Relative Areal Requirements for Bedded Salt Disposal of LWR Wastes from Several Fuel Cycles," private conversation, 1977.
- 6-53 Burkholder, H.C., "Nuclear Waste Partitioning Incentives," (BNWL), paper presented at U.S. NRC Technical Workshop on Waste Partitioning, Seattle, Washington, June 8-10, 1976. See also Burkholder, H.C., "Methods and Data for Predicting Nuclide Migration in Geologic Media," paper presented at International Symposium on the Management of Wastes from the LWR Fuel Cycle, Denver, Colorado, July 11-16, 1976.
- 6-54 Marsily, G.; Ledoux, E.; Barbreau, A.; and Margat, J., "Nuclear Waste Disposal: Can the Geologist Guarantee Isolation?" Science, 5 August 1977, Volume 197, No. 4303, pp. 519-527.

SECTION 7

STATUS AND ISSUES OF SYSTEM REQUIREMENTS

7.1 INTRODUCTION

The previous two sections have described in detail the process, development history, and major issues concerning most of the major components of two distinct systems for handling the back-end of the fuel cycle. These two waste management systems are intended for the disposal of two distinct forms of high level waste-unreprocessed spent fuel, and high level waste from a reprocessing plant. This chapter will examine the issues concerning the integration of these components into their respective systems. The reason that the system-level issues are important is because it is only at this level of analysis that the problems of the capability of the components to handle the demands made on the entire system (system inputs) can be examined; because the disposal of the actual system inputs is the problem in need of a solution.

One last introductory remark concerns the overlap of component objectives in the spent fuel storage/disposal system. The independent spent fuel pool, the SURFF, and geologic storage are all designed to handle spent fuel on an interim basis. It is doubtful that any one reactor would have all of its fuel transferred sequentially to each one of these facilities. However, during this initial development period for the back-end of the fuel cycle, it is possible that the public sector along with the private sector may develop various combinations of the components in different regions of the country. Such a variety of constructed facilities would provide actual data to make comparisons for future decision. Because of the likelihood of the use of each of the overlapping alternatives, it was considered important to include each of these in the component level discussion. In order to provide a useful parallel to the High-Level Waste System discussion, all of the components in the spent fuel storage/disposal system will be integrated.

7.2 PROCESS DESCRIPTION

The following is a list of the important system factors which are discussed in the paper.

- Transportation
- Facilities required
- Major R&D program required
- System cost and ownership
- Institutional responsibility

A discussion and comparison of each of these factors follows below.

7.2.1 Transportation

In Section 3 a simplified projection of the annual number of shipments was presented for the growth of nuclear generating capacity to 380 GWe. Figure 4-6 showed projections for (1) all shipments by truck; (2) all shipments by rail; and (3) shipments by a mixture of truck and rail. The problem of transportation is examined in Reference 7-1. To quote from the main findings of that report:

"Increased concern over the safety of shipments of spent fuel and high-level waste has been expressed by state and local governments, by railroads, and by organized intervenor groups. Some state and local governments have attempted to restrict or prohibit the shipment of radioactive materials through their regions. ERDA has challenged these actions on the basis of federal preemption of the regulations of such shipments. Resolution of these cases adds uncertainty to the cost of the existing facilities and the siting of future nuclear facilities. The American Association of Railroads believes that special trains with speed limited to 35 miles per hour that stop when other trains pass by are required for shipping of nuclear materials. If a train were to carry spent fuel rods, it would not be permitted to carry any other freight. The proposed tariff charges that the railroads have filed before the Interstate Commerce Commission have been protested by ERDA and the nuclear industry. ERDA and the nuclear industry have challenged the railroads' position that they are common carriers of nuclear materials. The ICC has allowed the proposed tariff charges to go into effect pending a final decision on their propriety. Under this new tariff, all rail shipments of nuclear spent fuel and waste must be moved only in special trains, at a cost in addition to the regular freight charge of about 20 dollars per mile, or about 30,000 dollars each for a 1,500 mile one-way trip."

"In contrast the position of the trucking industry appears to be enthusiastic about shipping nuclear materials. However, the extensive use of trucks could lead to increased costs of unloading the spent fuel rods from the reactor spent fuel pools. The truck spent fuel shipping casks hold one fuel assembly, while the rail casks (IF-300) hold 7 assemblies. In addition, the increased number of shipments could have an adverse impact on the public acceptability of these shipments. According to the ERC Nuclear Sub-Committee staff, "If a combination of economic and regulatory constraints make all present transportation modes infeasible or unacceptable, the possibility of Federal control of all shipping of nuclear waste may have to be considered. Concern over theft or diversion of fissile materials has already led to discussion of this possibility, and the Federal government has recently taken over all shipments of nuclear materials in the Department of Defense programs." (Reference 7-1, Vol. I pages 31 & 32)

When comparing spent fuel storage/disposal to HLW disposal, the difference in transportation requirements is dependent on the location of facilities with respect to reactor and other fuel cycle facilities. For example, if a spent fuel disposal-packaging facility were colocated with the ultimate disposal site, then the external transportation requirement from the packaging plant to the disposal site would be eliminated. The low mileage impact that these shipments would have on the total rail and track shipment in the U.S. (as shown in Section 4.2.3) holds for any case.

One variable that needs to be considered is the supply and demand for shipping casks since there are now only 6 truck casks and 4 train casks available (Reference 7-2). The number of shipping casks required depends on the projected growth in number of LWR's, the amount of spent fuel per cask, and the turn-around time per shipment. For example, if each truck cask contains 1 PWR assembly, or about 0.4 MTU of spent fuel, and current operating practice of 42 shipments per truck per year is followed, then each cask is capable of transporting approximately 17 MTU of spent fuel per year from a reactor site to a geologic repository. If reprocessing is delayed indefinitely, the total LWR capacity in the U.S. may reach a level of about 300 GW(e), and then remain virtually constant for 25-30 years thereafter. In that case about 9,000 MTU per year of spent fuel would be discharged and about 530 casks are adequate to ship this spent fuel to a repository. If production of these casks were to begin in 1980 at a rate of about 50 casks per year, the supply would keep pace with demand by the time the Federal repository is available. A larger number of shipping casks would be required if spent fuel is shipped first to a centralized or regional storage facility, or to a SURFF, and then transported later to a Federal repository for permanent disposal.

7.2.2 Facilities Required

Reactor Storage Capability

In ERDA 77-25 (Reference 7-3, pg. 11) the number of U.S. LWR's that would require additional spent fuel storage capability to permit scheduled discharges is estimated for the period 1977-1986. As stated in that report (page 5), "Where only preliminary planning* was indicated, no expansion was considered in this report." It is useful to look first at the estimates contained in ERDA 77-25, and then analyze the nature of the problem confronting the utilities. The ERDA estimates were made for three cases:

Case 1 corresponds to currently available facilities, including re-racking of existing spent fuel pools at reactor sites, and expansion "as planned" (early 1977), plus the GE Morris facility at 700 MT. Case 2 is simply Case 1 plus additional storage capacity of 1100 MT in 1980. Case 3 is the most optimistic and anticipates the additional storage of 1100 MTU at the General Electric Morris facility (as requested on April 30, 1977), and the availability of an Exxon storage basin of 3500 MTU in 1982 with additional expansion up to 7000 MTU as needed.

*by the utilities

For Case 1 the ERDA projections show that by 1986 27 of the 179 operating reactors would have utilized their full fuel core reserve and additional storage capacity, and would not be able to permit scheduled discharge. This conclusion is the reason for the observation by the Nuclear Assurance Corporation that:

"The utilities and AFR storage vendors indicate a high degree of awareness of the spent fuel disposition situation, and both industry segments are now taking positive action to assure sufficient spent fuel storage capacity." (Reference 8-3, page 5).

These ERDA estimates are based on the first re-racking designs (1974-1976), which did not make optimum utilization of available storage space because of the assumption that reprocessing, although delayed, would come "on line" in the early 1980's. Current designs take into account the decision to delay reprocessing indefinitely. These designs utilize all space to the maximum possible extent consistent with the constraints imposed by cooling requirements and noncriticality. Typical high-density racks design now provide for 3 or greater core loads, plus one full core load in reserve, or approximately 10 years of storage, as compared with 4 years of storage for the early designs.

If no decision on fuel reprocessing is made in the next few years the utilities will have two choices:

- (1) Transfer of spent fuel from one pool to another.
- (2) Construct independent storage pools either on-site or away from the reactor (AFR).

The decisions will be made on a utility-by-utility basis.

7.2.3 R&D Program

The estimated FY 78 budget for the ERDA Fuel Cycle R&D program showed \$33,400,000 for LWR Fuel Reprocessing, \$71,050,000 for Terminal Storage R&D and \$35,500,000 for waste processing, R&D and demonstration. Budget estimates for the recently initiated work on SURFF have not yet been determined. However, it is unlikely that the R&D required for SURFF would exceed the R&D that would be needed for LWR Fuel Reprocessing and waste processing. This is mostly due to the amount of existing research completed under the Retrievable Surface Storage Facility (RSSF) program. The \$71 million for terminal storage R&D is unaffected by the decision on reprocessing, since a program for geologic disposal is similar, especially in the near term, for HLW disposal and spent fuel disposal.

7.2.4 Cost and Ownership

When examining the cost of the back-end of the fuel cycle, the additional mills per kilowatt is found to be quite small. This follows from the large number of kilowatt hours that can be produced by nuclear

reactors. Consider the low growth scenario. The cumulative kilowatt hours produced at an assumed 60% capacity factor between 1976 and 2000 is 21×10^{12} Kw-hrs. If we assume a total waste disposal system cost of as high as \$10 billion, the additional cost per kilowatt hour is 0.46 mills. In comparison, the average bus bar generating cost for the nuclear reactor operated by Commonwealth Edison was 14.2 mills per kilowatt-hr in 1976. The cost of the back-end of the fuel cycle would represent an increment of 3 percent. However, this level of analysis can be misleading. One needs to look at the individual or micro-level decisions that are being made. Such an economic analysis is beyond the scope of this document. However one important aspect that demonstrates the difference between considered here is the ownership of the fissile material.

Under the HLW disposal system the ownership of the radioactive materials was fairly well defined. At the reactor, the utility owned the fuel. During the transportation of the spent fuel to the reprocessing plant the spent fuel was owned by the utilities, but the transportation company was liable for any injury caused by the radioactive material. (This phase would be covered by the Price Anderson Act). Under this plan upon arrival at the reprocessing plant the utility buys the reprocessing service. Incorporated in the price would be some credit for the extracted fissile materials. Also incorporated in the cost are the costs of processing the waste, storing the waste, transporting the waste and final disposal of the waste. The material at the reprocessing plant presumably would be owned by the reprocessors, until it is received at a Federal repository. At that time, present regulations state that the NRC will become the owner of the HLW. NRC has indicated that there is a typographical error in the present regulations and that the intended owner of the HLW is DOE.

Under the Spent Fuel Storage/Disposal options, several ownership schemes are possible. At this time, there are no firm plans as to the eventual ownership scheme. As an example, one could imagine the utilities paying for the storage and disposal cost at the time it first ships its spent fuel from the reactor to a storage facility, with a provision of a credit for the fissile material if reprocessing is eventually approved. Because of the uncertainty in the eventual decisions on reprocessing, the determination of an equitable storage/disposal fee would be complicated. As an alternative ownership example, the utilities could retain ownership of the spent fuel for as long as they wish, while paying a rental fee to the storage facility. Here again, the uncertainty in national spent fuel policy will make the determination of an acceptable storage fee difficult.

7.2.5 Institutional Aspects

The institutional aspects of nuclear waste management are quite complex. An exploratory study of these institutional constraints has been performed by JPL (Ref. 7-4). This brief discussion of ownership and storage fee determination is closely related to other institutional factors. At the industry level, the prospect of no reprocessing has not been a welcome concept. However, as Reference 7-3 pg. 1 points out, the "survey (of LWR spent fuel disposition capabilities) clearly indicates that utilities and the utility serving industry recognize that aggressive actions are required to insure sufficient spent fuel storage, and have identified plans

to provide the needed storage." This observation was made prior to the President Carter's April 7, 1977 decision to indefinitely defer reprocessing.

The utilities face other decisions in addition to establishing adequate storage until Federal actions provide a longer term solution. One is their own determination of the value of holding on to the spent fuel. Related to this is the determination of whether the spent fuel on hand can be considered an asset, because of the potential fuel content, or a liability, because of the future costs of storage and disposal. Such decisions will probably not be made just on the basis of economic efficiency. The public relations aspect of declaring spent fuel a waste, will also be taken into account by the utilities. The licensing and public hearings phases of the expansion of spent fuel storage capabilities at reactor sites are also important factors in decision-making. The differences between the two systems appear to be significant. However, most of the problem and uncertainty are due to the transition from the reprocessing model to an interim storage model.

This uncertainty is responsible for many of the well defined and some poorly defined problems at the Federal level. Reference 7-1 describes some likely institutional interactions that could take place during the licensing of the facilities required for HLW disposal. Similar types of interaction will exist between the Environmental Protection Agency (setting of standards for high-level waste), the Nuclear Regulatory Commission (criteria setting, approvals, and licensing) and the Energy Research and Development Administration (site selection, program definition, design, construction, and operation) for a Federally operated central pool, SURFF or a geologic repository. However, all existing programs in each of these bureaucracies must now be modified to account for the possibility of spent fuel storage and disposal. At this time it is too soon to establish how quickly this can be done in comparison to the ongoing development work for the HLW disposal system.

7.3 FLOWCHARTS OF U.S. MAINLINE PROGRAM FOR COMMERCIAL HIGH-LEVEL NUCLEAR WASTES

The flowchart displayed in Figure 7-1 is presented to provide guidance in determining the interactions between technology activities and the approval cycle to develop a waste disposal system. Figure 7-1 is divided into two main sections of flow. The upper section indicates high-level waste repository development and approval activities. The lower section indicates treatment and packaging technology development and approval activities.

The principal Federal interactions between EPA, ERDA, and NRC are shown in this figure. This diagram represents a procedure requiring quite strong, formal regulatory action by NRC. To date the NRC has not decided whether to require either a formal site review or a construction license before ERDA builds the high-level waste repository. In the portion of the diagram for solidification and packaging, the assumption has been made that this plant is to be built as a private facility; hence it requires NRC licensing. If the plant were built as an ERDA facility, present law does not require a NRC license.

An analysis of the HLW management programs and plans is provided in Reference 7-1. This analysis was performed for the Office of Science and Technology Policy specifically examines broad approval categories of public policy and acceptability, programs and plans to identify strengths and inconsistencies, schedule implications, and needed work activities. On the subject of interagency coordination, this report makes the following observations:

"The resources devoted to high-level waste management by NRC and EPA have been inadequate in view of the critical roles these agencies play in developing standards, criteria and regulations. The budgets of these two agencies have recently been increased substantially. However, we question whether the resources available to these two agencies are commensurate with their responsibilities. There appears to be a lack of an adequate platform for systematic discussion of the sufficiency of the scientific data base for geologic disposal. An important outcome of such discussion would be the development of decisions as to what parameters should be monitored, and for how long, during the test phase of a specific repository site. In addition, there appears to be a need for improved coordination of the nation's high-level nuclear waste program."

To operate a repository for commercial high-level nuclear waste by 1985, a complex set of activities must be undertaken by the Federal government. An Ad Hoc Interagency Task Force on Commercial Nuclear Waste Management was formed in the spring of 1976 (Ref. 7-6) to ensure internal compatibility of the Federal government's activities in this area. The task force was chaired by OMB and included representatives from CEQ, EPA, ERDA, NRC, and USGS. This task force was phased out at the end of July 1976 (Ref. 7-7) and replaced by the White House FRI Committee (Ref. 7-8, pg. 112). The FRI Committee provided information for President Gerald Ford's Nuclear Policy Statement on October 28, 1976. At present, there appears to be no interagency committee for coordinating the Federal government's efforts in commercial nuclear waste management, although such a group is called for in the National Energy Plan (Ref. 7-9, pg. 72).

A JPL modification of the OMB interagency task force flow chart showing the interconnection of the government's high-level waste management activities is presented in Figure 7-2. This chart shows considerably more detail than Figure 7-1, which focused principally on regulatory activities. The major milestone activities are high-lighted with ellipses at the top of the chart. These major activities include providing a better definition of high-level waste; determining regulations for interim storage; preparing environmental standards for high-level waste disposal; development of general regulations by NRC; docketing of the environmental reports and applications, and construction of the repository; obtaining an operating license; and finally, receiving the high-level waste at the national repository. A similar flow chart should be prepared for spent fuel disposal.

The sub-activities which must be accomplished in order to lead to each major activity are shown below the respective major activities. Alternate sets of sub-activities have been shaded in order to make the

chart easier to read. The following abbreviations have been used in Figure 7-2:

TAD	Technical Alternatives Document
DGEIS	Draft Generic Environmental Impact Statement
EIS	Environmental Impact Statement
INT	International
ENV	Environment
ACRS	Advisory Committee on Reactor Safeguards
DEIS	Draft Environmental Impact Statement
FEIS	Final Environmental Impact Statement
SER	Safety Evaluation Report
FSAR	Final Safety Analysis Report
FSER	Final Safety Evaluation Report.

Figure 7-2 shows that the majority of the activities occur during 1977 and 1978. In order to ascertain the credibility of the proposed schedule, and also to determine if there are major technological or regulatory gaps in the proposed schedule, it is necessary to examine the nuclear waste programs of EPA, NRC, and ERDA. A detailed examination of the available material from EPA, NRC, and ERDA is presented in Appendices B, C & D of reference 7-1.

7.3.1 Timing of the Development of Required Facilities

The development of the facilities required for the back-end of the fuel cycle has had a history of poor performance as described in each of the component sections in Sections 5 and 6. Whether or not a SUREF or a geologic repository will be available by 1985 is still an open question. The comparison between the two systems discussed in this chapter is difficult to make at this time because of the relative newness and lack of preciseness in the definition of the spent fuel storage and disposal system.

Therefore, any attempt at comparison of the availability of the two systems would be inappropriate. One reason is the often quoted statement that the further away a technology is from actual industrial development the better it looks and vice versa. The HLW option has been studied for many years. Research and Development on the long-term spent fuel storage and disposal option has begun only recently. Though this study has not been able to find enough established research to make the comparison, it can be noted that the 1985 date appears to us to be unrealistic for either high level waste management option.

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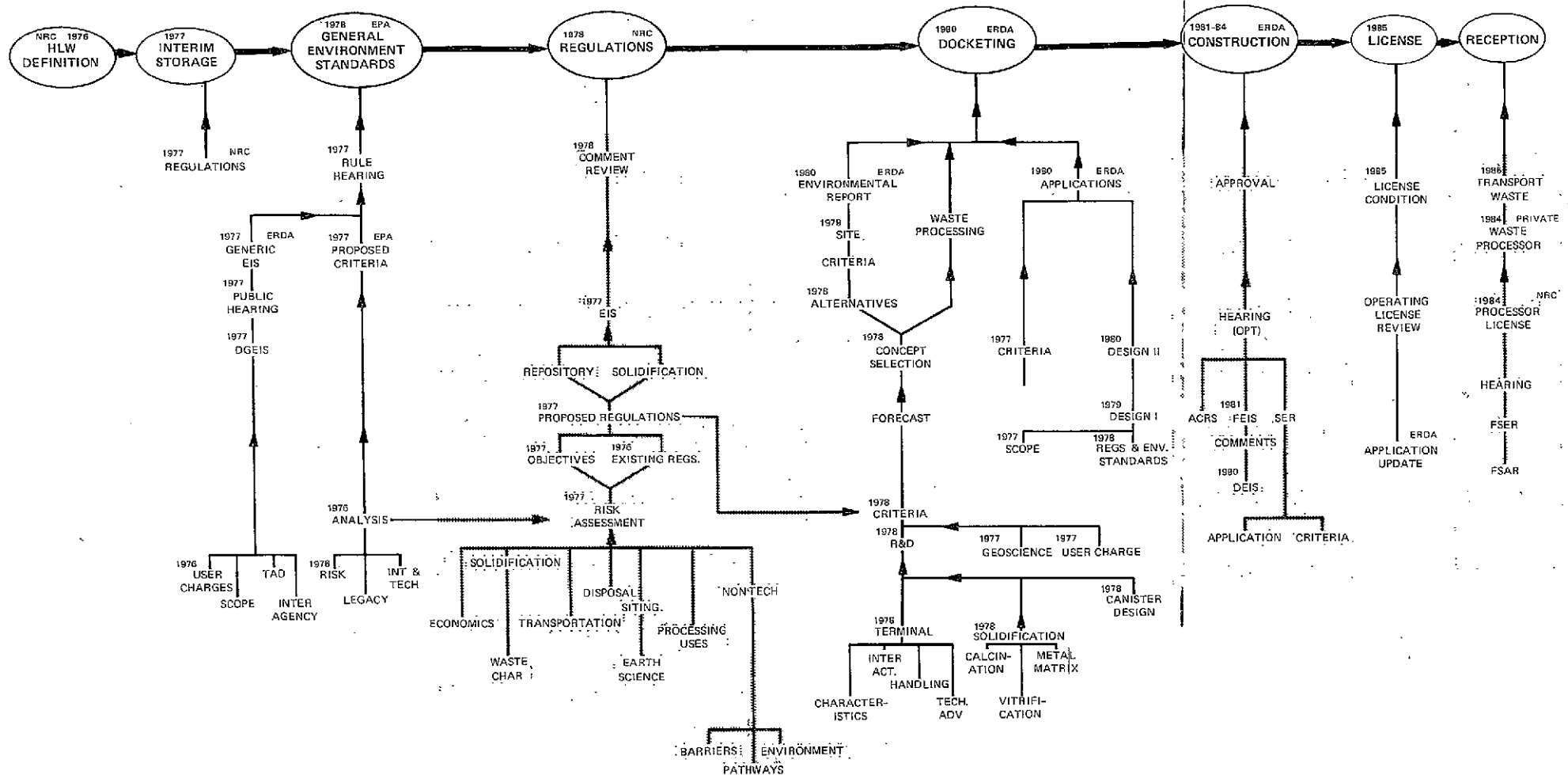


Figure 7-2. Timing Chart for High-Level Nuclear Waste Management

7.4 SECTION 7 REFERENCES

- 7-1 English, T.D., et al., An Analysis of the Back End of the Nuclear Fuel Cycle with Emphasis on High Level Waste Management, JPL publication 77-59, Volumes I and II, Jet Propulsion Laboratory, Pasadena, California, August 12, 1977.
- 7-2 Jefferson, R.M. and Bonzon, L.L., "Available Packaging and Transportation Systems," SAND76 5544, Sandia Laboratories, Albuquerque, New Mexico 87115.
- 7-3 ERDA, LWR Spent Fuel Disposition Capabilities, 1977-1986 (ERDA 77-25), May 1977.
- 7-4. Kuehn, T. J., An Exploratory Study of the Social and Institutional Constraints of Nuclear Waste Management, JPL Publication 77-45, Draft, Sept 15, 1977.
- 7-5. Federal Decisions in the Nuclear Fuel Cycle, Draft #2, Federal Resources Council, 1977.
- 7-6. Memo from J. Lynn, OMB; to R. Train, EPA; R. Peterson, CEQ; R. Seamans, ERDA; and W. Anders, NRC; March 25, 1976.
- 7-7. Telephone conversation between R. Weiher, Chairman of Ad Hoc Interagency Task Force on Commercial Nuclear Waste Management, and T. English, JPL, April 14, 1977.
- 7-8. Kuhlman, ERCDC testimony, March 24, 1977, p. 112.
- 7-9. National Energy Plan, Executive Office of the President, Energy Policy and Planning, April 29, 1977, p. 72.

SECTION 8

CONCLUSIONS

In the preceding sections of this report we have examined the technical status of two major U. S. programs for high-level nuclear waste management. Each major component of both the "old mainline program" for HLW from reprocessing plants, and the newly-developing program for disposal of unprocessed spent fuel has been analyzed for its Scientific Feasibility, Technical Achievability, and Engineering Achievability. Systems aspects common to both programs, such as transportation and logistics, also have been examined. Our principal findings and observations are summarized in this Section.

8.1 "OLD MAINLINE HLW" PROGRAM

Even though calcination, vitrification, and packaging of the HLW are planned to be operated as an integrated process, the technical status of each of these specific steps is discussed separately. Because the concept of geologic isolation in bedded salt is common to both the HLW and spent fuel systems, it is discussed in Paragraph 8.3.

8.1.1 Interim Liquid Containment of High-Level Waste

The three issues of concern in this area were: (1) stability of the liquid during interim containment, (2) compatibility of the acid liquid waste and the steel tanks, (3) availability of adequate control technology to take care of off-gases coming from the tank. Based on an extensive review of the literature and conclusion that the experiences at INEL were pertinent, these issues are now considered to be resolved. Specifically, there appears to be no significant change in the composition of the liquid waste, over time, that would affect future operations. The history of storage at INEL and measurements from test-coupons in the tank indicate that liquid acid-waste and the existing steel storage tank are compatible for interim storage. The filter and monitoring control systems have been adequate to handle the off-gases.

8.1.2 Solidification of the Liquid in a Spray Calciner

The Scientific Feasibility of the spray calciner concerns its ability to produce a solid, dry waste form that is chemically, radio-lytically, and thermally stable. The spray calciner has been developed in connection with an in-can melter, and as a result the calcine is immediately converted to glass in this system. Since there is zero holdup time in the spray chamber, the waste is in a solid, dry (fine powder), calcine form only for the period necessary for gravity-feeding it into the melter and to attain fusion of the calcine glass frit mixture. Therefore, the stability issue should be addressed to the final product of the spray calciner/in-can melter, namely, the vitrified waste (see Paragraph 8.1.3).

Technical Achievability of the spray calciner depends largely on the reliability of the nozzle that atomizes the incoming liquid waste fuel. A pneumatic atomizing nozzle with a ceramic air cap is now commercially available, and has shown minor wear (0.01 mm wear on a 6.3 mm-nozzle orifice after 1000 operating hours). This remotely replaceable nozzle is a significant improvement over the earlier troublesome metallic atomizing nozzle, and is expected to remain in service for several months of operation. The incoming liquid waste concentration is not a matter of concern, because there is no reliance on physical contact of the waste with a hot surface. Drying of the waste is accomplished by a combination of radiative and convective heat transfer.

So far as Engineering Achievability is concerned, spray calcination has been under development at the Battelle Northwest Laboratories (BNWL) for over 16 years. A full scale non-radioactive facility has been operating since April 1977, and has calcined at a rate of up to 200 l/hr. A full scale remote operation and maintenance non-radioactive facility is now in design that will allow experience to be gained in remote canister handling and maintenance techniques. Radioactive tests using high-level waste from commercial irradiated fuel will be carried out in late 1978 utilizing a pilot plant spray calciner/in-can melter process (15 l/hr). Based on this program and the experience gained in the earlier WSEP program between 1966 and 1970 with fully-radioactive engineering scale operational runs, engineering achievability of this technique appears likely.

8.1.3 Immobilization of Solidified Waste by Vitrification

Principal effects that determine the Scientific Feasibility of incorporating the calcined waste in a composite borosilicate glass are as follows: (a) radiation effects; (b) thermal effects; including thermally induced stresses and devitrification; and (c) mechanical shock.

The principal radiation effects on the glass will be produced by α -radiation; β and γ radiation cause relatively minor amounts of damage, and neutrons and fission recoils produce only minor damage because of their low fluxes in the waste. Experiments with Curium-244 "doped" glass which have attempted to simulate 3,000 and 50,000 year exposure times show that the stored energy in zinc borosilicate glass will probably not exceed 50 cal/gram. The adiabatic release of this stored energy would lead to temperature increases in the glass not exceeding 200°C. Such a release of stored energy (at 0.08 cal/°C gm) could possibly result only from a rapid rise in the waste's temperature caused by an external heat source or lack of cooling source. Similar results were obtained with both lead borosilicate and high-silica glasses. The effects of α -radiation on glass density and helium generation also appear to be small. The effects of α -radiation on long-term glass leach rates, which are determined by experiments with potassium-doped glass, exhibited increases in leachability by about a factor of 2.

Thermal shock effects will be produced in the glass as a consequence of the thermal stresses caused by both the initial cooling of the solidifying glass and by the decreasing thermal gradient in the canister

with time as the "decay heat" of the fission products decreases (secondary cracking). The extent of this "secondary cracking" which does occur will be a function of initial thermal power in the canister, the fin and canister design, the storage conditions, and the position of the glass within the canister. Also, the thermal shock sensitivity will be dependent on the thermal conductivity, thermal expansion coefficient, and heat capacity of the glass. An example of potential stressing factors might be the cracking effect due to variations in the thermal expansion coefficients of unmelted feedstock and the surrounding glass. Tests to examine this and other factors on a simulated long time scale are planned by BNWL for 1978. The initial cracking will result from the free-air cooling of the canister at a rate of $400^{\circ}\text{C}/\text{hour}$ and produce an increase in the surface area of the glass by a factor of 10.

Devitrification, or crystal formation, in the zinc borosilicate glass can be a problem at temperatures of the order of $700\text{--}750^{\circ}\text{C}$. Simulated long-term leach test results indicate that devitrification effects at 700°C increase the leachability by a factor of 10. The leach rate of some simulated samples has been initially observed to be as low as $10\text{--}7\text{ gm}/\text{cm}^2/\text{day}$. Devitrification effects reduce the time interval required for slowly moving water to leach 10% of the glass from about 4×10^4 years to about 4×10^3 years.

The Technical Achievability of the waste-glass is being investigated by means of an empirical approach in order to establish guidelines for composition, processing techniques and handling. A considerable amount of information has been obtained for a reference glass composition with 28 wt.% radioactive waste, and investigations have begun on three other waste compositions representative of the range expected in the discharge from various types of reprocessing plants. A level of waste oxides of 23 wt.% has been determined to be the optimum considering processing factors (such as volatility, homogeneity, and corrosion), leachability and economic factors.

The most important aspect of Engineering Achievability concerns the potential difficulty of remote maintenance and the need for safe remote operations. Vitrification tests of selected, important remote features are currently under way at BNWL on a scale of $150\text{ l}/\text{hr}$ (nearly full scale). Next, a full-scale non-radioactive remote operation and maintenance cell is planned to be constructed. Pilot plant remote demonstration tests at a scale of $15\text{ l}/\text{hr}$ are planned to be carried out in a "hot-cell" using commercial high-level waste.

8.1.4 Encapsulation

The major factors in establishing the Scientific Feasibility of the encapsulation process are: (1) material problems during melting, including internal and external corrosion of the canister, and deformation as a result of metal creep; and (2) temperature fluctuations in the

canister wall caused by air cooling, water cooling, or by removal from a water storage basin, during handling operations. Because of the lower thermal expansion coefficient of the encapsulated glass, thermal stresses can cause some fracturing of the glass. Also stress corrosion can occur in the canister. Stress corrosion in water can be minimized if the chloride content of the water is less than 1 ppm and the pH is alkaline. Other materials being examined besides the current 304L stainless steel are 310 and 347 stainless steel, and Inconel alloys 600, 800, and 802.

Technical and Engineering Achievability of the encapsulation stage depend sensitively on the containment criteria for both processing and long-term storage conditions. Canisters made of 304L and 310 stainless steel were filled with various waste-glass compositions during the WSEP program, and stored in a water basin, an air basin, and in ambient air storage. After about 2 1/2 years of storage no significant changes caused by radiation were found, and the canisters are still in storage. The current WFP program is examining canister corrosion under both processing and long-term storage conditions. From the various designs and materials under study, it appears that a canister can be developed that will provide the relatively short-term containment needed for interim retrievability.

The conclusions pertinent to disposal of both reprocessed HLW and unprocessed spent fuel are presented together in section 8.3, since many of the considerations are similar.

8.2 DISPOSITION OF UNREPROCESSED SPENT FUEL

8.2.1 Spent Fuel Storage at Reactor Sites

A spent fuel capacity of 1 1/3 "reactor cores" for a single 1000 MW(e) reactor was considered adequate in the 1960s, in order to provide a full-core reserve storage in the event of emergency situations, or maintenance and inspection, even if one year's discharge (1/3 core) was already in the pool. Because of the delays in developing fuel reprocessing plants in the early 1970's, and especially the 1977 Presidential decision to deter reprocessing indefinitely, re-racking, or "densification" of the storage pools became essential. The important technical issues here are "criticality", cooling, and seismic effects.

New high-density fuel storage racks have been designed and constructed by each nuclear reactor vendor for their specific reactor fuel as well as by several independent companies for various customers (e.g., Exxon Nuclear-designed racks for Rancho Seco). The use of more accurate criticality codes, and fixed poisons in the racks has reduced the centerline-to-centerline distance of the fuel in the racks for PWR assemblies from a previous value of about 21 inches to about 12 inches. For BWR high-density racks, the reduction has been from about 10 inches to 8 inches.

As a result, it is now feasible to increase the storage capacity of an existing reactor pool from 1 1/3 - 1 2/3 reactor cores up to 3 - 4 cores by re-racking. Allowing for a full-core reserve, this

expansion would permit up to 9 years of storage capacity for the storage of spent fuel for a single reactor.

As of July 1, 1977, 16 nuclear reactors had obtained NRC approval for re-racking, and an additional 18 reactors had submitted requests for approval to NRC. This constitutes a total of 34 reactors out of 63 operating nuclear plants in the U.S. These re-racking requests vary from a 30% to a 250% increase in spent fuel pool capacity, with a median of 120%. If all 34 reactors expand their spent fuel storage capacity by re-racking to a capacity of 4 full cores, the total available storage capacity in excess of a full core reserve would amount to 10,200 MTU. As of December 31, 1977, the cumulative amount of spent fuel discharged is expected to total 2,650 MTU, of which about 1,000 MTU can be stored at the GE facility at Morris, Illinois and at NFS, West Valley, New York. Thus, 8,700 MTU capacity would be available for future discharge, or the equivalent of 6 year cumulative discharge for all 63 reactors operating at 70% load factor. This estimate assumes that NRC will approve all 18 pending requests for re-racking and will also permit intra-utility and inter-utility transfers, if necessary.

8.2.2 Independent Off-Site or On-Site Spent Fuel Storage

If the decision to reprocess spent fuel is deferred indefinitely, the utilities will be faced with some difficult choices in the next few years. Possible options include: (1) Provide additional on-site spent fuel pool storage; (2) Join in a private consortium to construct independent off-site spent fuel pool storage, or contract with potential builders and operators of such facilities; (3) Declare the spent fuel as "waste" and call upon the Federal government to accept it. Obviously, these decisions will be strongly influenced by economic factors. For example, the cost of storage is now estimated at \$10,000 per MT in 1977 dollars, so that the annual cost of storing spent fuel in water pools increases by about \$330,000 per year. Over a ten-year period the cumulative cost is \$18.5 million, which amounts to about 0.3 mills/kwh in 1977 dollars. In order to decide whether it could be profitable to continue storing spent fuel, these costs have to be compared with the estimated value of the spent fuel, should reprocessing be approved at some future date.

While this report was in preparation, Exxon announced (August 1977) that it was not even seriously considering building spent fuel storage capacity without a decision to go forward with reprocessing. Mr. Ray Dickeman, President of Exxon Nuclear, stated that "I just did not consider such an alternative to be responsive to this nation's long-term energy needs."* Thus, either the utilities could build such capacity themselves, or the Federal government might have to assume responsibility for construction and operation of interim off-site spent fuel storage facilities as insurance against the possibility of unforeseen delays in the 1985 operational schedule for the Federal repository in bedded salt. This decision by the Federal government may also be strongly influenced by

*Nuclear Fuel, August 8, 1977, page 8

the need to provide storage capability for foreign reactors operating on enriched uranium fuel provided by the U.S.

In addition to these economic and international political factors, there are several important technical factors to be considered in designing a spent fuel storage facility so as to minimize radiation hazards to the public. These include: (1) direct radiation from the irradiated fuel; (2) liquid release to the environs from either an excessive storage basin leak or low activity storage vault leak; (3) airborne release from a potential fuel cladding rupture or cooler leak.

Direct radiation could occur only if the water level in the basin is too low to provide adequate shielding because of excessive evaporation (cooling system failure), or excessive basin leakage, and at the same time, the make-up water capability is inadequate. The water depth would also be inadequate to provide shielding in the event of a "fuel drop" accident that might result in the formation of a critical array. For the case where both basins at the Morris, Illinois, G.E. facility are filled (700 MTU of spent fuel), the maximum water temperature is calculated to be 85°C (185°F), and it would require at least 9 days to evaporate the more than $39,000\text{ ft}^3$ of water before the tops of the fuel rods would be exposed. The temperature would rise at a rate less than 2°F/hr for about 3 days, and more slowly thereafter. Because multiple backup sources of water are provided, there is a very small probability that there would not be sufficient make-up water available in the event of a temporary drop in storage basin water level.

Conceivably, inadvertant drainage of the spent fuel basin could also be caused by penetration of the stainless steel liner. The floor of the unloading basin shelf and deep pit include provisions to dissipate impact loads from a maximum cask drop accident. The shelf has a fabricated, stainless steel crushable finned pad and a 2-inch steel plate on top of the stainless steel liner. The floor of the pit is covered by a $1\frac{3}{4}$ -inch steel plate under the stainless steel liner to dissipate the impact load from a 125-ton shipping cask falling 31 feet.

The strength of the cask fins will result in the total energy of the drop being absorbed by the pad. The finned pad on the unloading shelf (two 1-inch stainless steel plates separated by 4-inch, $\frac{1}{2}$ -inch thick steel fins) is expected to absorb all of the cask's energy. For a drop on the shelf edge or pit floor, the load on the shelf concrete is less than its ultimate dynamic load and the stress is expected to be less than ultimate stress of the floor concrete due to the distribution of the load over a relatively large area.

If the concrete were to fail, penetration is expected to be less than an inch and the stresses on a flat plate that is deflected an inch are well within the elastic stress range for steel. The conclusions are that the floor and shelf of the unloading pit are adequately protected from a postulated IF-300 cask fall.

So far as liquid release to the environs is concerned, the structure is equipped with a leak detection system and pump-out system to handle potential leaks. The low activity vault is capable of withstanding

a design basis earthquake. Also, no heavy objects are moved over the vault. There is no credible accident that could rupture the tank and concrete sufficiently to release liquid radioactive material to the environs.

The two possible paths for airborne release of radioactive gases are: (1) rupture of the fuel cladding; (2) leakage of the cooling system heat exchanger. The fuel cladding could be ruptured by dropping a fuel bundle or assembly, or by dropping a loaded fuel basket containing 4 PWR bundles or 9 BWR bundles. Both of these occurrences are low probability events. In the first case, if all the fuel rods in a bundle are ruptured, releasing the fission gases to the basin water, and all released fission gases are expelled from the stack over a period of 2 hours, the maximum resulting radiation levels have been calculated to be 10^{-1} mrem whole body dose (for BWR) and 1 mrem thyroid (PWR). In the second case, if a fuel basket fell from the maximum height of 22.5 feet above the basin floor, and all the fuel rods in all the bundles were ruptured (an extremely unlikely event) the maximum off-site exposures would be about 80 mrem. So far as cooling system leakage is concerned, even a postulated leak rate as large as 1.5 grams per minute would result in maximum release rates of 10^{-3} mrem/hour whole body dose and 7×10^{-4} mrem/hour for the thyroid dose.

8.2.3 Engineering Interim Surface or Shallow Sub-Surface Storage (SURFF Storage)

If a Federal repository for deep, geological disposal of spent fuel were not available by 1985, unpackaged spent fuel assemblies at a reactor site or in a centralized storage basin might be taken to an engineered spent unprocessed fuel storage facility (SURFF). This facility would place the fuel assemblies in a metal container to isolate the fuel rod cladding from the external environment, and then store them retrievably in an area that provides biological shielding from the radiation produced by the spent fuel.

At the present time, SURFF is a "concept" that is being developed for further consideration, rather than a formal "program". The similarities between SURFF and the old RSSF for high-level waste storage are obvious. The three passive systems studied under the RSSF program are also candidates for SURFF, namely (1) Sealed Storage-Cask; (2) Air-Cooled Vault; (3) Storage in Near-Surface Heat Sink. The Committee on Radioactive Waste Management (CRWM) of the NAS-NRC in its review of the RSSF concepts in 1975 recommended that an optimized version of the Sealed Storage Cask Concept be pursued. However, in this same report the Committee listed nine areas where "final specifications and designs have not been determined and considerable additional engineering R&D are necessary." Seven of these areas apply to SURFF. In the absence of program definition and specification of design criteria by ERDA, no conclusions can be drawn at the present time about the Technical or Engineering Achievability of SURFF.

8.2.4 Packaging of Unreprocessed Spent Fuel for Disposal

Studies of techniques for packaging unreprocessed spent fuel for disposal are in a preliminary stage. This status is due largely to the relatively recent emphasis placed on this technology by President Carter's April 7, 1977 statement indefinitely deferring commercial reprocessing. In the absence of clear program definition and specification of design criteria no conclusions can be drawn at the present time about the technical achievability or engineering achievability of the technology for packaging unreprocessed spent fuel for disposal.

8.3 GEOLOGIC DISPOSAL OF ENCAPSULATED HIGH-LEVEL WASTE AND UNREPROCESSED SPENT FUEL; DEEP GEOLOGIC ISOLATION IN BEDDED SALT

8.3.1 General Conclusions

From the time of the first report in 1957 of an advisory committee of the NAS-NRC¹ up to the present, deep geologic isolation of nuclear wastes in bedded salt has been recognized as the most promising method for containment of the radioactivity in these wastes over periods of time as long as a million years. The experiments conducted in Project Salt Vault demonstrated that the technical and engineering problems associated with the safe handling of highly radioactive materials in an underground environment, the behavior of the bedded salt in response to the radioactivity and decay heat emitted by the encapsulated waste (or spent fuel), and the structural design of the underground repository, are all solvable with currently available technology. Differences of opinion about the risks inherent in this method of waste disposal therefore center (or rather, should center) around the long-term integrity of the repository, taking into account man's intrusion into the salt bed during construction, testing and monitoring. The principal open questions are concerned with the possible intrusion of circulating water in significant quantities, followed by dissolution of a portion of the salt in the disposal layer, and subsequent geologic transport of radionuclides to the biosphere and thence to man.

8.3.2 Technical Aspects of Repository Design: Packaging, Canister Spacing and Temperature Effects on Bedded Salt

During the first 100 years of storage, the total radioactivity of the spent fuel is generated mainly by the fission products, and differs only slightly (15-20% higher) from the total radioactivity in the high-level waste after reprocessing. A similar conclusion naturally follows for the "decay" heat rate. Thus, for the first 100 years, the containment problem for the spent fuel in bedded salt is not much different in principle from the problem for encapsulated HLW.

For storage times longer than 100 years, the time histories of radioactivity and decay heat rate for spent fuel and for HLW differ significantly. The decay heat rate (per metric ton of uranium) is more than 10 times higher for spent fuel than for HLW over the period 10³ - 10⁵ years after discharge from the reactor. To maintain the integrated heat

load for spent fuel and HLW at the same levels over the first 10^4 years, the initial areal power density in the disposal layer for spent fuel must be 1/5 that for HLW. This power density is achieved by packaging a single spent fuel assembly in each canister and spacing the canisters about 10 meters apart. Of course the number of canisters required per GW(e)-year is increased from 15 for HLW to 75 for spent fuel. Present 1-D models show temperature rises of only $10^\circ - 15^\circ \text{C}$ in the "far-field" at a distance of 500 m above the disposal layer after 10^4 years of storage. The phenomena associated with the long-term response of bedded salt and surrounding earth and rock to thermal gradients requires careful study using a more realistic 3-D model.

The most important design parameters for the "near-field" are the canister spacing and the duration of pre-storage, or "aging" before emplacement in the repository. For example, if each canister contains a single spent fuel assembly and the pre-storage time is 10 years, the maximum temperature rise at the canister wall (ΔT_{MAX}) is 35°C for a canister spacing of 10 meters. ΔT_{MAX} is 60°C if the pre-storage time is 5 years. For HLW the maximum temperature rise at the canister wall with a canister spacing of 10 meters is 170°C if the pre-storage time, T_0 , is 5 years and 100°C if the pre-storage time is 10 years, based on an areal power density of 100 kw/acre or 20 watts/m². Except for HLW with $T_0 = 5$ years, these maximum temperature rises are well below the levels at which any significant difficulties are expected because of salt decrepitation or brine migration in the salt toward regions of higher temperature (or higher salt solubility). The sensitivity of ΔT_{MAX} to the design parameters of pre-storage time and canister spacing is great enough so that the designer can readily select these parameters to meet any conceivable restrictions on ΔT_{MAX} .

For the "open" LWR nuclear fuel cycle, the total number of canisters and the size of the repository (or repositories) required for containment and isolation of spent fuel over the entire life of the LWR can be estimated on the basis of these design considerations. Recent estimates of total U.S. reserves of U_3O_8 made by a special panel of the National Academy of Sciences places these reserves (including known, probable, possible and suspected) at about 2×10^6 tons. Using current technology this reserve would produce 3×10^5 tons of UO_2 fuel enriched to a ^{235}U content of 3.5%. If the single fuel assembly packaging alternative were adopted, about 7×10^5 canisters would be required to contain this amount of UO_2 . Suppose these canisters are stored in 8 separate repositories. In each repository the canisters could be arranged in a square array (plan view) with a spacing of 10 meters between canisters. The entire array would occupy an area 3×10^3 meters by 3×10^3 meters or 9 square kilometers (about 3 square miles) or about 2000 acres.

8.3.3 Long-Term Repository Integrity

Obviously, it is impossible to demonstrate the long-term (10^6 years) integrity of a bedded salt repository experimentally. Therefore, current and past geologic evidence and analytical techniques must be utilized to establish the range of probabilities for the escape of some significant amounts of radioactivity from the repository. At the

present time the weakest links in the site evaluation and selection process are as follows:

- (1) Specification of hydrogeological criteria necessary to ensure site stability;
- (2) Understanding, measurement, and modeling of the current ground water regime in any given area of interest;
- (3) Extrapolation of that ground water regime up to one million years into the future;
- (4) Understanding of geospheric transport mechanisms, and especially the sorption factors for the long-lived radionuclides in the actual geophysical environment.

Experience to date with the sites at Lyons, Kansas; Asse, West Germany; and Carlsbad, New Mexico, show that a number of "scenarios" can be constructed in which a connection between the repository and the biosphere is established. These include:

- (1) Natural dissolution of the salt beds by the action of circulating ground water;
- (2) Establishment of hydraulic connections between the overlying aquifer(s) and the disposal horizon by failure of mine shaft or borehole sealing, or faulting and displacement upward of the disposal horizon;
- (3) Establishment of hydraulic connection between overlying and underlying aquifers resulting from failure of borehole plugging or faulting.

The estimated probability of each of these scenarios is closely related to the inadequacy of the data base required in steps (1)-(4) listed above. Sandia has attempted to produce a reasonably detailed assessment of the probabilities and consequences of such events for the WIPP site near Carlsbad, New Mexico. If ground water flow through the repository does in fact take place, the most important index of the effectiveness of the geologic barrier for any particular radionuclide is the ratio of the geospheric transport time to the half-life of the nuclide ($T_{1/2}$). This transport time depends upon two parameters: (1) the "quality" of the geologic formation, as expressed by the mean pore velocity of the mobile phase; (2) the "hold-up" factor, K, or ratio of mean pore velocity to the nuclide migration velocity. This factor K depends on the degree of sorption of the radionuclide by the processes of ion exchange, ion absorption, etc. The following significant conclusions for ^{239}Pu are drawn from simplified one-dimensional calculations:

- (1) Without sorption, even an excellent confining geologic formation only introduces a delay in the transport of the radionuclides to the environment; it does not retain it.

- (2) With a high degree of sorption ($K \cong 105$) even a relatively poor geologic formation retains the ^{239}Pu for a sufficiently long time so that radioactive decay prevents it from reaching the environment.

The degree of sorption depends critically on the complex geochemistry of plutonium. A comprehensive program of laboratory and field studies is required on the interactions of radionuclides with both individual minerals and rocks. Credible predictions of radionuclide behavior cannot be made without this information. Useful information can also be obtained by examining the behavior of radionuclides in the field, e.g., at the Hanford site, the Nevada weapons test site, and the Oklo, Gabon "natural reactor" site.

The Office of Waste Isolation has the responsibility for commissioning and evaluating a set of studies and experiments designed to answer some of these questions. Their studies to date include: (1) identification of salt formations of interest; (2) reconnaissance studies; (3) area studies. This work has not yet progressed to the next levels of detailed confirmation studies and in-situ tests. In addition, ERDA and NRC have not yet developed design criteria, site suitability criteria, and site selection criteria. Thus, extrapolation of existing information to predict Engineering Achievability is very difficult.

At the present time the Scientific Feasibility of deep geologic disposal of encapsulated high-level nuclear waste and unprocessed spent fuel in bedded salt has not yet been proven. However, there is no scientific reason to anticipate that existing, planned and recommended work will not create a scientific data base that will reduce the hydrogeological uncertainties to acceptable low levels, although the proposed time schedule may be too optimistic. The possibility of surrounding the repository with a man-made geochemical barrier (such as clay) should not be ruled out.

During the 10-20 year period of testing, monitoring, data analysis and projections, the geologic repository will necessarily be operated in a retrievable mode. This procedure has two important favorable by-products: (1) By storing spent fuel in the repository the pressure for continual expansion of on-site or independent off-site spent fuel pool capacity is considerably reduced, if not eliminated, and the need for SURFF may also be eliminated; (2) If a decision to proceed with reprocessing should be made during this period, spent fuel could be retrieved and transported to a fuel reprocessing plant. The solidified high-level waste discharged from the FRP could be encapsulated in canisters designed to fit into the storage vaults and emplacements previously occupied by the spent fuel canisters. The discussion of major design parameters for the repository in Sec. 8.3.2. shows that such a plan is well within the capabilities of current technology.

8.4 Matrix Summary

In Section 3, a matrix method of summarizing the technological status of high level waste management systems was described. This method is applied in Figure 8-1 and 8-2, in order to visually summarize the findings of this report for both HLW from a reprocessing plant, and unprocessed spent fuel, respectively. The numbers shown in the matrix elements refer to sections in the conclusions where these findings are summarized.

8.5 SYSTEM REQUIREMENTS

Transportation system requirements produce issues relating to transportation facilities, R&D, institutional interactions, cost, and ownership. The "special train" issue is of great significance for transportation considerations. Trains carrying spent fuel or solidified high-level waste would be limited to the following constraints: One shipment is allowed per train; other freight and passengers are prohibited; a maximum speed of 35 mph is imposed; and the train is required to stop when other trains pass by. These constraints, coupled with the projected number of shipments, could cause significant problems to the entire rail network.

The number of spent fuel shipping casks required to service a 300 GW(e) nuclear power industry is estimated to be 530. If production of these were to begin at a rate of 50 casks per year in 1980, the supply would keep pace with demand by the time a federal repository for High-Level Waste is available. This cask production rate is feasible, although it necessitates a considerable increase in present fabrication rates.

Analysis by ERDA has indicated that the utilities may be faced with a shortage of spent fuel storage capacity which could significantly lead to the shut down of a significant number of reactors. The utilities and spent fuel storage vendors are aware of this potential problem and are taking corrective action through approaches such as high-density re-racking of spent fuel pools at the reactor, and also considering building centralized spent fuel storage at away-from-reactor sites.

The R&D program for disposal of high-level waste has been significantly impacted by President Carter's decision to indefinitely defer reprocessing. The "old mainline program" for solidified high-level waste is no longer viable. A new program is being developed for the permanent disposal of spent fuel. This program is in the early formative stages so it is difficult to make a detailed analysis of its credibility. However, statements by ERDA spokesmen that this program will meet the 1985 repository date appear overly optimistic.

Uncertainties regarding both the potential for commercial reprocessing and the economics of reprocessing complicate the issue of long-term ownership of spent fuel. If the utilities continue to store the spent fuel, they will bear the additional costs in the hopes of making some future economic gain. If, on the other hand, the utilities declare the spent fuel as waste, the government may have to bear these storage costs. This issue is presently unresolved.

Stage of Technology Development System Component	Scientific Feasibility	Technical Achievability	Engineering Achievability
Interim Liquid Containment Treatment and Packaging For Disposal • Calcination • Vitrification • Encapsulation Transportation Bedded Salt Emplacement and Disposal		8.1.1 8.1.2 8.1.3 8.1.4 8.3.3	8.1.1 8.1.2 8.1.3 8.1.4 8.5

Figure 8-1. Matrix Summary of the Technical Status of Components of the "Old Mainline" High-level Waste Disposal System

Stage of Technology Development System Component	Scientific Feasibility	Technical Achievability	Engineering Achievability
Storage • At Reactor • Away from Reactor - Wet • SURFF			8.2.1 8.2.2 8.2.3
Packaging & Treatment For Disposal			8.3
Transporation			8.5
Bedded Salt Emplacement and Disposal		8.3.3	

Figure 8-2. Matrix Summary of the Technical Status of Components of the Newly Developing Waste Disposal System for Unreprocessed Spent Fuel

APPENDIX A

ORIGIN, NATURE & DISPOSAL OF HIGH LEVEL WASTE

A.1 SCOPE OF THIS APPENDIX

The purpose of this appendix is to provide a brief and simple account of the origin, nature and disposal of high level waste. It is a background paper and does not go into the detail of the main report and it does not justify or give references for every statement made. It is hoped that it may help a non-specialist reader to get an idea of the present state of knowledge and technology.

The appendix deals not only with high level waste in the waste stream from a reprocessing plant (IOCFR50 Appendix F), but also with other material of comparable radioactivity, such as spent fuel.

A.2 SOURCE AND NATURE OF HIGH-LEVEL WASTE

When a nuclear reactor is refueled, four types of radioactive materials are removed in the "spent fuel":

- (1) Fission products; these are a highly radioactive mixture of elements formed by the splitting of atoms of uranium and plutonium in the fuel. The radioactivity consists in the emission of β -particles (fast moving electrons) and γ -rays (energetic X-rays). The radioactivity of the fission products is discussed in Section A-5 below.
- (2) Unburnt uranium which was part of the fuel with which the reactor was charged.
- (3) Transuranic elements; these are elements heavier than uranium they are formed by the addition of neutrons to atoms of the fuel. The principal elements present are neptunium (Np), plutonium (Pu), americium (Am), and curium (Cm). If the reactor fuel included plutonium, some of it would remain in the spent fuel in addition to that formed in the reactor. Many of the isotopes of the transuranic elements emit α -particles (fast moving helium nuclei) and γ -rays, others emit β -particles and γ -rays. The radioactivity of the transuramics is discussed in Section A-5 below.
- (4) Radioactivity induced by neutron bombardment in structural materials, particularly in the tubes in which the fuel is contained. The radioactivity of this material is qualitatively similar to that of the fission products but the amount of radioactivity is less by a factor of about 100.

When the spent fuel is removed from a reactor, it is contained in tubes made of Zircaloy (an alloy containing 98% of zirconium). These "fuel rods" vary from one reactor to another; in the Pressurized Water

Reactor (PWR), which is the commonest type of Light Water Reactor (LWR), they are 3.85 m long and 1.0 cm in diameter (Figure A-1). The rods are arranged in groups called "assemblies"; in a PWR each assembly contains 264 rods, it is 4.06 m long and 22 x 22 cm in cross section (Figure A-2). In a Boiling Water Reactor (BWR) the rods are larger and there are only 64 of them in an assembly, the assemblies are smaller in cross section than those of a PWR.

The spent fuel is removed from the reactor as assemblies. These assemblies are initially stored, just as they come from the reactor, in pools of water at the reactor sites. Owing to uncertainties in the rate of construction of reactors, the prediction of the number of assemblies to go into storage in the future is subject to great uncertainty. Recent estimates are much below those current a few years ago (compare, for example, References A-2 and A-3). The EDRA estimate in Reference A-3 is, perhaps as good as any. It was made in August 1976 and predicts that by the year 2000 there will be a nuclear generating capacity of 510 GW3 in the U.S. (the estimate may well turn out to be high by as much as a factor of 2 but, for planning purposes, it is important not to under-estimate the problems). The load-factor of these stations is estimated to be 55% averaged over their 40-year life, but in the year 2000, most of them will be in their prime and are expected to have a load factor of 70%. If these prophecies prove correct, the annual production of spent fuel in that year would be 34,000 assemblies (Reference A-3, page 17) containing about 5 million fuel rods and weighing 15,000 metric tonnes (all tons in this appendix are metric tonnes, a metric tonne is 1000 kg or 1.1 short tons of 2000 lb; the distinction is of no importance here and a metric tonne will be called a ton throughout). In 1976, 2400 assemblies of spent fuel weighing 940 tons were discharged.

If all the reactors were LWRs fueled with slightly enriched uranium (3.3% U-235) and no plutonium, then 15,000 tons of spent fuel would contain:

<u>Substance</u>	<u>Amount (tons)</u>
Uranium oxide (about 0.8% U-235)	11,320
Fission products	400
Plutonium oxide	100
Other transuranic elements	16
Structural materials	<u>3,164</u>
	15,000

If the reactors were fueled with a mixture of uranium oxide and 1.5% plutonium oxide, the amount of plutonium oxide in the 15,000 tons of spent fuel would be about 190 tons instead of the 100 tons for fuel containing uranium oxide only.

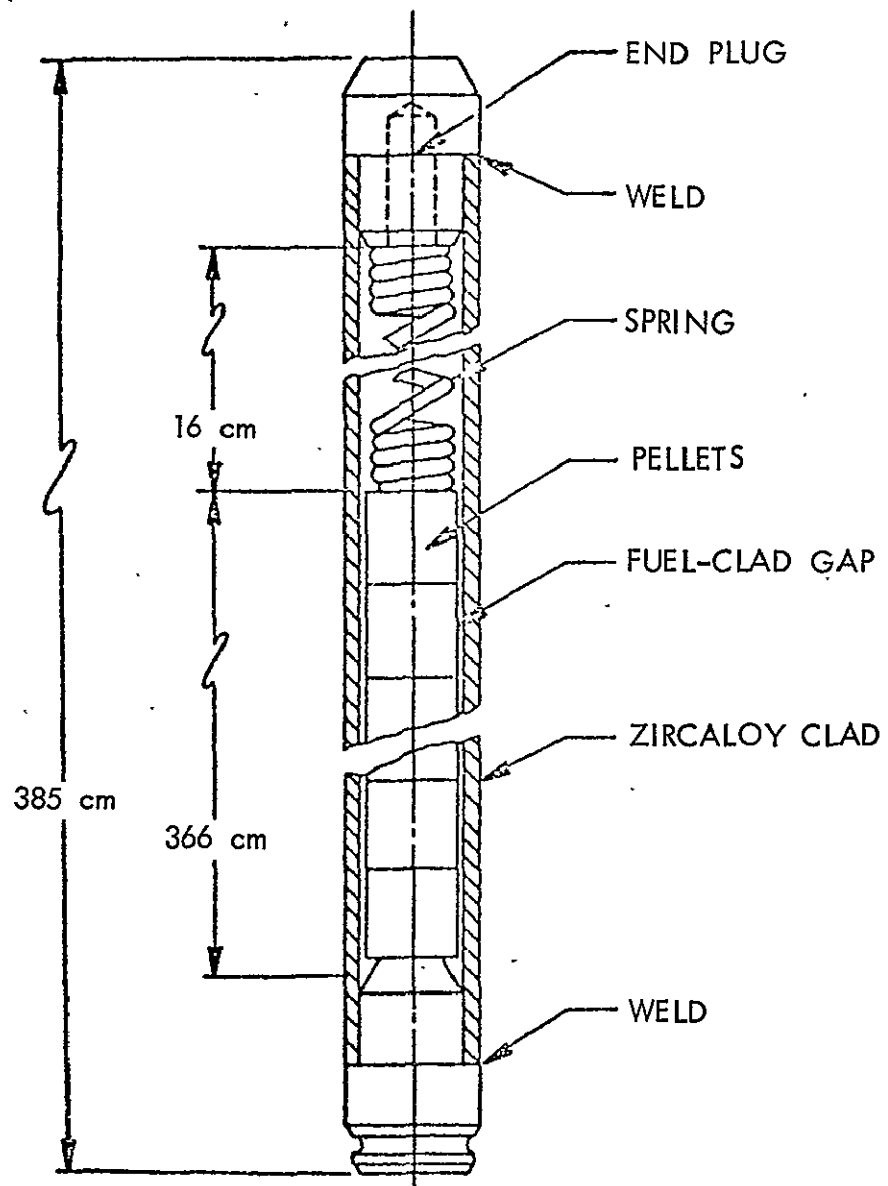


Figure A-1. PWR Fuel Rod (from Reference A-1)

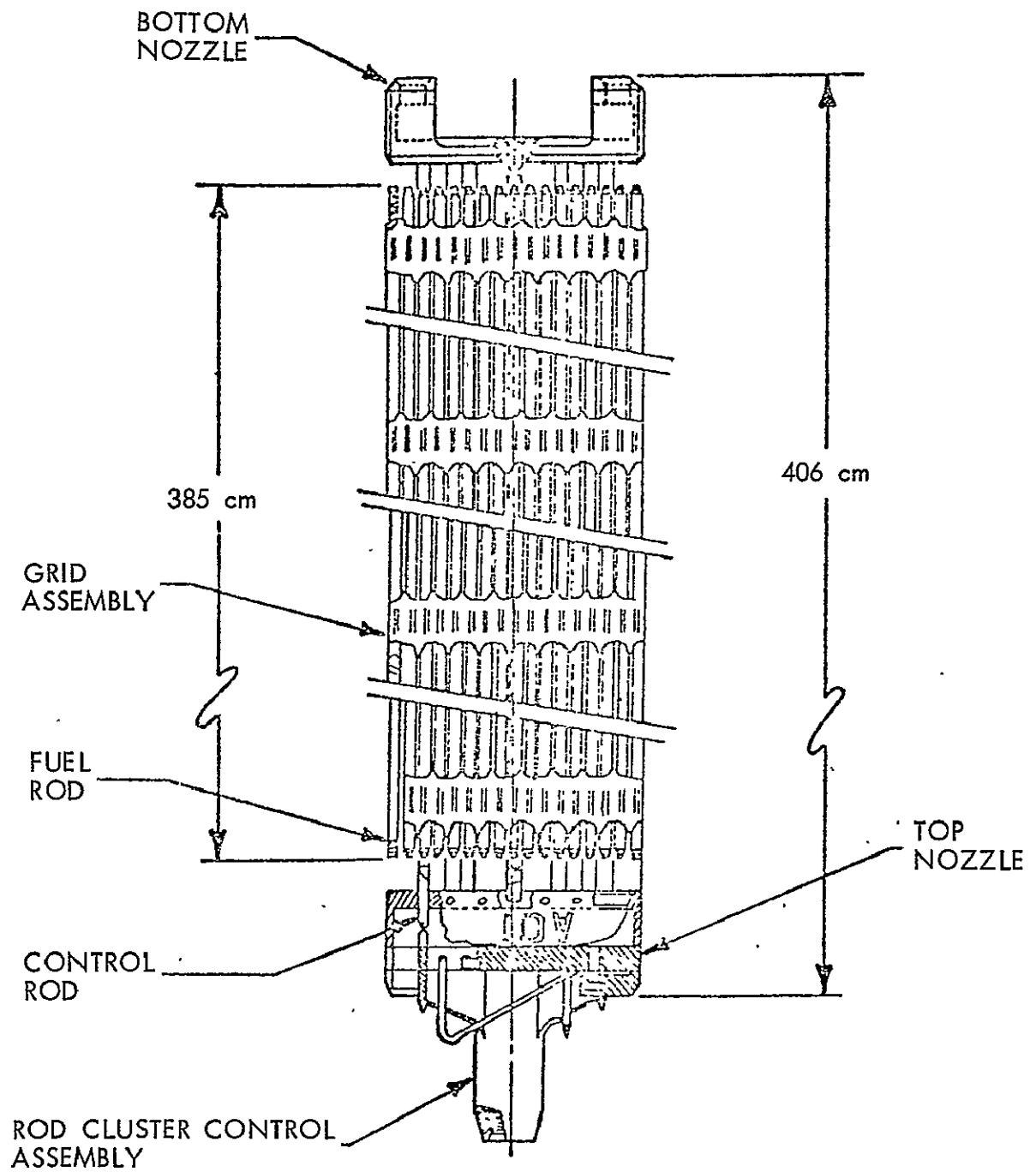


Figure A-2. .PWR Fuel Assembly (from Reference A-1)

A.3 TEMPORARY STORAGE AT REACTOR SITES

When the spent fuel is removed from a reactor, it is too radioactive to be transported and must be stored in a pool of cooling water at the reactor site. When these pools were designed it was envisioned that the fuel would be removed after 160 days and transported to a reprocessing plant. In fact there are, at present, no reprocessing plants for the spent fuel from commercial power reactors, and the spent fuel is accumulating at the reactor sites. Further, the President has decided that reprocessing will be indefinitely postponed.

A decision as to what to do with the spent fuel is needed since, if nothing is done, before 1985 31 reactors will be forced to close (Reference A-4). A larger number will not have enough space to store their fuel if it were necessary to unload the whole core (the provision of storage space for this is a condition in the licenses of some, but not all, reactors).

There are two possibilities:

- (1) To extend the storage at reactor sites by enlarging the pool's or packing the assemblies closer together (additional neutron absorbing materials are then needed to prevent the pool becoming a reactor). The spent fuel could then be kept in the pools until either reprocessing is started or geological storage is available. The earliest date for either of these events is 1985 and, in view of the slowness of the licensing process, considerable delay beyond this is possible. The accumulation by 1985 has been estimated as (Reference A-3, page 17) 58,000 assemblies and by 1990 as 128,000 assemblies. A decision is needed as to whether so much storage distributed at reactor sites is acceptable from the point of view of possible deterioration of the fuel rods or of possible terrorism.
- (2) To limit storage at reactor sites and remove the spent fuel to protected storage at a few centralized highly protected sites either under federal control or operated by private industry. Since the spent fuel at these storage sites would be several years old, it might be possible to dispense with water cooling.

A.4 REPROCESS, STOW AWAY, THROW AWAY

If the President's decision to postpone reprocessing indefinitely became a permanent ban, then the spent fuel must be regarded as waste and permanent storage must be provided for it. This is the "throw away" option. The option involves the loss of about 98% of the energy potentially available from the fuel if both reprocessing and breeders were operating.

Until it is clear that not to reprocess is a fixed U.S. policy, it would be prudent to arrange storage in such a way that the spent

fuel could be recovered from storage for reprocessing. This is the "stowaway" option. The ability to recover is, in any case, desirable during the development phase of permanent storage. Storage would, presumably, be in containers, each holding one or more assemblies of fuel rods.

If, at some time in the future, the decision not to reprocess were reversed and reprocessing were restarted, the accumulated spent fuel would be sent to reprocessing plants and emerge as a nitric acid solution containing the fission products, about 0.5% of the uranium and plutonium and almost all the other transuranic isotopes (the removal of neptunium would be practicable, the removal of americium and curium is more difficult and no operable process at present exists). The uranium and plutonium (or perhaps the uranium only) would be re-used as reactor fuel. The solution of fission products, actinides, and the residual 0.5% of fuel would be solidified. The solidified high level waste would be mixed with suitable inert ingredients and melted to a glass in a metal can. These cans would be the form in which the high-level waste would be stored.

If the 15,000 tons of spent fuel considered above were reprocessed, the waste would emerge as a glass containing:

<u>Substance</u>	<u>Amount (tons)</u>
Uranium oxide	57
Fission products	230
Plutonium oxide	0.5
Other transuranic elements	16
Glass making oxides (SiO ₂ , B ₂ O ₃ etc.)	<u>900</u>
Total	≈ 1,200

A reprocessing plant also produces other radioactive waste streams which must be carefully managed. These are:

- (1) The chopped zircaloy tubes from the fuel rods; these hulls would be washed, packaged and stored with the high-level waste.
- (2) Gaseous and volatile fission products which are released from the fuel rods when they are dissolved. The important ones are Krypton-85, Iodine and Tritium. Krypton can be stored in gas cylinders or absorbed in zeolites; iodine and tritium can be converted to solid compounds and stored.
- (3) Filters, ion-exchange resins, contaminated plant, and trash from the reprocessing plant. These could be concentrated and put with the high-level waste or buried in shallow trenches according to the degree of hazard involved.

In addition to the spent fuel and waste from commercial reactors, there is a large quantity of high level waste derived from reprocessing

the spent fuel from reactors operated to produce plutonium for military purposes. This waste is contained in tanks at Hanford, Washington; Savannah River, South Carolina; and Idaho Falls, Idaho. The waste at Hanford is contained, for the most part, in mild steel tanks--a number of which have leaked. Double walled stainless steel tanks have proved satisfactory. This military waste is being solidified. Most of the uranium, plutonium and part of the cesium and strontium have been removed and are stored separately (see Reference 6; Table following page 109 for details). The material originally consisted of about 760,000 m³ of liquid. When all is solidified it will weigh about 400,000 tons. The disposal of this waste will not be discussed in any detail in this paper; there are severe problems, but they arise from choices made in the past, and the matter has little relevance to waste disposal from present or future commercial power programs, except as a warning of the degree of care needed.

The problem of the disposal of commercial waste is essentially to prevent the radioactivity of spent fuel or of the solidified high-level waste from a reprocessing plant from coming into contact with people during its period of toxicity. In practice this implies the protection of the whole biosphere. There is no possibility of absolute protection, some detectable quantity of radioactivity from every phase of the fuel cycle may appear in air and water. It is therefore necessary to define acceptable limits to the effluents, and to ensure that the arrangements for the disposal of spent fuel or solidified high-level waste, from a reprocessing plant stay within these limits. The limits set by NRC will be found in IOCFR20 Appendix B, Table 2 (see also IOCFR20.106). The matter is further discussed in Section A-5 below.

A.5 RADIOACTIVITY OF SPENT FUEL AND WASTE

In a discussion of the radioactivity of spent fuel or solidified high level waste, it is desirable to consider the fission products and the heavy elements (that is uranium plus the transuranics and their decay products) separately since their chemical and biological properties are very different, as also is the decline of their radioactivity with time.

The fission products are the main contributors to the radioactivity of waste and spent fuel for the first 100 years after it is produced. Initially, the fission products are a complex mixture of many different radioactive isotopes. Many of these have short decay times and, after 10 years, 92% of the radioactivity is in the isotopes strontium-90 and cesium-137 and their immediate decay products. After 1000 years, these are reduced to a negligible level. There still remain minute quantities of a few long-lived fission product isotopes, such as technetium-99 but their radioactivity is less than that of the transuranic elements, even in reprocessed fuel. Table A-1 lists the main contributors after 10 years of cooling.

The radioactivity of the transuranic elements (of which plutonium is one) is initially less than that of the fission products but, since it decays more slowly, it becomes the predominant contribution for all times over 600 years for reprocessed waste. For unprocessed spent fuel, plutonium is the main contributor to the radioactivity after

TABLE A-1

Radioactivity of fission products in spent fuel from 1 tonne of U 10.5 years after removal from a Pressurized Water Reactor (U enriched to 3.2% U-235, 30 MW/ton, 33000 MWd/ton). All isotope contributing more than 0.1% to the total curies from fission products are included (from reference A-2 Table 4-2, Pg. 18).

Isotope	1/2 Life	Curies
H-3	12.3y	403
Kr-85	10.8y	5960
[Sr-90	28.1y	60700
[Y-90	64 hr	60700
[Pd-106	367d	550
[Rh-106	130m	550
Sb-125	2.7y	678
Cs-134	2.05y	8380
[Cs-137	30.0y	85600
[Ba-137	2.6m	80000
Pm-147	2.62y	7750
Sm-151	87y	1160
En-154	16y	4530
Total		318000

The short period isotopes in the bracketed pairs are daughters of the long period ones and decay with them. For other isotopes and times see references A-2 and A-3.

200 years. Reprocessing reduces the amount of uranium and plutonium without much affecting the other transuranic elements.

For a general view of the variation of radioactivity with time, it is convenient to consider the decline in the number of atoms disintegrating per second without reference to their chemical nature or to the type of decay or energy release involved. Radioactivity, in this sense, is measured in curies. One curie of radioactivity is 3.7×10^{10} disintegrations per second (this apparently arbitrary number has an historical origin; it is the number of disintegrations per second in a gram of radium). The variation in the radioactivity of fission products and of the heavy elements in reprocessed waste is shown in Figure A-3. Unfortunately, no reliable figures for the radioactivity of unprocessed spent fuel for times exceeding 10 years are available to the writer. The radioactivity of the fission products in spent fuel will be close to that in the reprocessed waste shown in Figure A-3. Since 99.5% of the uranium and plutonium are assumed to be removed in reprocessing, the radioactivity of the heavy elements is greater in spent fuel than in reprocessed waste. After 10 years the difference is a factor of 30.

TABLE A-2

Radioactivity of transuranic elements in spent fuel from 1 tonne of U at 10.5 years after removal from a reactor (conditions as for Table A-1). All isotopes giving more than 0.1% of the total curies from transuranic elements are included (from Tables 4-3, Pg. 20 and 5-8, Pg. 54 of Reference A-2).

Isotope	1/2 Life	Curies at 10 yr
Pa-233	27d	-
Np-237	2.1 x 10 ⁶ y	-
Np-239	2.4d	-
Pu-238	86y	2700
Pu-239	24400y	323
Pu-240	6580y	479
Pu-241	13.2y	65300
Am-241	458y	1410
Am-242	152y	-
Am-243	7950y	-
Cm-244	17.6y	1670
Cm-245	9300y	-

By the year 2000, the annual addition to the high-level waste is estimated to be 1250 million curies and the amount in storage, after allowing for decay, to be 5900 million curies (Reference A-3). These estimates depend on the estimate of future nuclear power production and assume reprocessing. The radioactivity of spent fuel, if there were no reprocessing, would be slightly greater.

The radioactivity of the accumulated military waste is said to be 590 million curies of which 150 million has been separated and encapsulated (Reference A-6; Table 3 following page 109). A hundred thousand curies of cesium and strontium and 650kg of plutonium have been intentionally buried or accidentally spilled. At present, the radioactivity of this military waste exceeds that of the accumulated commercial waste. After the mid-1980s, the radioactivity of the commercial spent fuel and waste will exceed that of the military waste. This discussion does not include the radioactivity of stored military spent fuel.

Radioactivity stated in curies is a poor measure of biological effects. It is therefore necessary to consider the other physical and biological factors that influence the biological changes produced by radioactivity.

A.6 BIOLOGICAL EFFECTS OF RADIOACTIVITY

The radioactivity of nuclear waste measured in curies is known but is insufficient to estimate its biological effects, or to set standards for the amount that can be allowed to escape into the air,

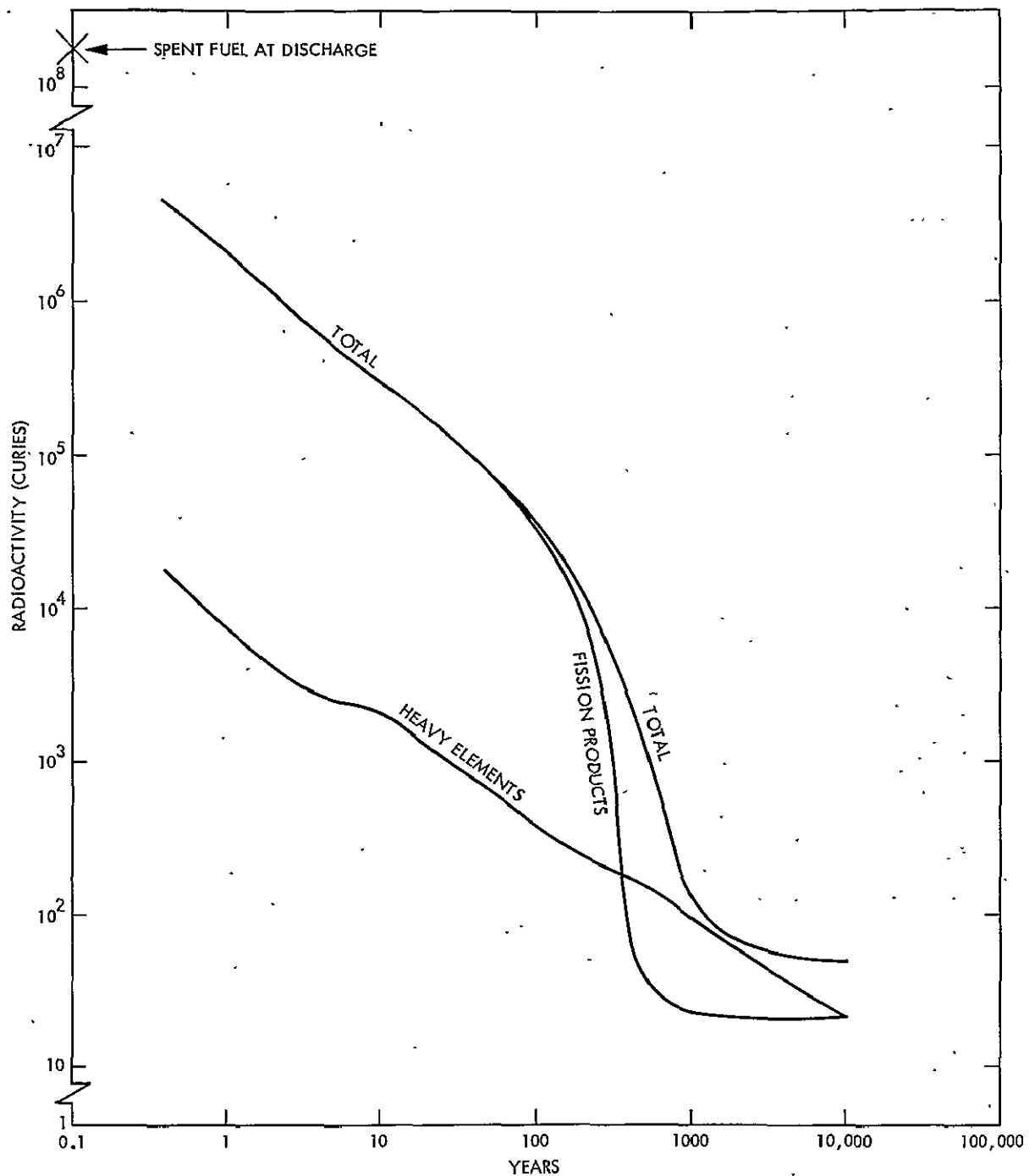


Figure A-3. Radioactivity of Re-processed Waste derived from 1 ton of Uranium Fuel. Note: The zero of time is the time of withdrawal of spent fuel from the reactor. The radioactivity of spent fuel at this time is shown on the left-hand axis. (from Reference A-2).

rivers, or the ground. It is necessary to consider various scenarios in which radioactive material comes in contact with animals or human beings. Typical scenarios are:

- (1) Radioactive Krypton-85, which is a gas, escapes into the air, is dispersed by the wind and is breathed into the lungs.
- (2) Ground water is contaminated by coming in contact with buried waste, it emerges from the ground and gets into a river and thus into a water supply.
- (3) Particles of waste are blown by the wind into a field, they settle on the grass and are eaten by cows.

The examples illustrate two important features of the biological effects. The krypton in the first example is chemically inert, it will get into the blood stream from the lungs and will become uniformly spread through the body. In the course of a few days or weeks, it will work its way out through the lungs or in urine. Whether the result is likely to be serious depends on the amount of radioactive material breathed, but it is at any rate better to have it make only a temporary stay than to have it settle permanently in the body.

The strontium-90 eaten by a cow will, in large part, stay permanently in her body. It is chemically similar to calcium and most of it will settle in bone and remain there for the rest of the cow's life. Another part will go, again with calcium, into the cow's milk. If this is drunk by a baby it will go into the baby's bones, which are particularly liable to radiation damage. This is an example in which a radioactive material passes up the food chain and whatever organism is at the top of the chain gets an unexpectedly large dose.

Clearly, the effects of the escape of radioactive waste depend in a complicated way not only on the physiology of a single animal, but on the whole network of relations between all living things. Even if all this were completely understood, which it is not at the present time, there still remains the problem of specifying what dose is acceptable.

It is not possible here to go into the details of this very complicated question an admirable account will be found in Reference 5. The present regulations governing the amount of each isotope allowed in air or water will be found in 10CFR20 (see especially 20.4 for units, 20.101 for doses and Appendix B to section 20 for allowed amounts in air and water). The regulations for waste disposal sites have not yet been issued by the Nuclear Regulatory Commission, but may be expected to follow the provisions of 10CFR20 and to require licenses issued under conditions similar to those of 10CFR50 which apply to reactors. In its response to a draft of this report EPA has indicated that "...although EPA has not yet proposed an environmental radiological protection standard for management of HLW it is unlikely that the standard when promulgated will permit radiation doses as large as those permitted by 10CFR Part 20."

The system of allowable doses of radiation is based on energy absorbed in tissue. The unit of absorbed energy is the rad which is defined as 0.01 Joule/kg. Since the biological effect depends not only on the amount of energy absorbed but also on the type of radiation, it is usual not to use the rad directly but a unit called the rem. The number of rems is obtained by multiplying the absorbed dose in rads by a factor to account for differences in biological effectiveness due to the quality of radiation and its spatial distribution in the body. The factor is unity for X-rays, α -particles and γ -rays, and 10 for β -particles (this last factor is not clearly stated in the regulations). The allowable dose averaged over the whole body of an individual may not exceed 5 rem/year for a person working in a "restricted place" (i.e. a nuclear facility such as a reactor building) or 0.5 rem/year for a person in an unrestricted area. The Environmental Protection Agency (EPA) specifies in 40 CFR 190 that the environmental standards of the uranium fuel cycle operations shall provide reasonable assurance that: "The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations." All these doses are additional to the inevitable dose due to cosmic rays and the natural radioactivity of the ground, water and air. The natural dose varies from place to place, in the USA the range is 0.07 to 0.22 rem/year. The actual doses at the boundary of a reactor site are normally less than 1% of the EPA limit for the general public.

In addition to the general limits on whole-body dose, there are limits to the amount of each radioactive isotope allowed in air and in water flowing out of a restricted area. These limits have been chosen so as to take account of the tendency of certain isotopes to concentrate in particular organs (e.g., iodine in the thyroid, strontium in bone) and also of the possibilities of concentration in food chains. A few examples (from 10CFR20 Appendix B) are given in Table A-3 as an illustration of the magnitudes involved. The extremely low limit set for plutonium in air is noteworthy, at this limit the plutonium in a litre of air would give only one disintegration in 5 days.

This elaborate system of limits is not based on detailed experiments on human subjects and obviously could not be. The belief that it represents a satisfactory safeguard against radiation-produced cancer or harmful mutations is based on:

- (1) The fact that the dose in rem to the whole body and to critical organs is less than the natural dose,
- (2) The experience of the Hiroshima survivors, and the results of accidental exposure of human beings to radiation, particularly
- (3) The results of animal experiments

Table A-3. Amounts of a few radioactive isotopes allowed in air and water for unrestricted use (from 10CFR20 Appendix B)

Isotope	Air Ci/m	Water Ci/m
Strontium-90	3×10^{-11}	3×10^{-8}
Iodine-129	2×10^{-11}	3×10^{-8}
Cesium-137	2×10^{-9}	2×10^{-5}
Plutonium-239	6×10^{-14}	5×10^{-6}
Americium-241	2×10^{-13}	4×10^{-6}

NOTE

These are the limits for soluble compounds; different limits are set for insoluble compounds.

It is not practicable to make experiments at the low levels of interest in regulating contamination from nuclear facilities; the whole point of such regulations is to ensure that the number of cancers or genetic errors produced in a small, and preferably an undetectably small, proportion of those occurring naturally. To keep and store even a million mice is hardly practicable, and, in any case, forms only an indifferent model of the U.S. population. The results for very low doses are therefore deduced from those at higher doses by assuming a linear relation between dose and casualties.

The assumption of linearity has been attacked both by those who believe there is a threshold dose below which no harm is done by radiation and by those who believe low doses are proportionately worse than high ones. This question seems more likely to be resolved by the development of understanding of the processes of damage and repair through the study of molecular biology than by direct experimentation on animals at low doses. The report of the National Academy of Sciences (Reference 5) discusses the matter at great length and recommends the use of a linear law for extrapolation to low doses.

In addition to numerical limits for exposure of individuals and populations, and for contamination of air and water, the NRC regulations contain the provision (10CFR20.1) that "persons...should...make reasonable effort to maintain radiation exposures, and releases of radioactive materials in effluents to unrestricted areas, as low as is reasonably achievable". "Reasonably achievable" means "...taking into account the state of technology, and the economics of improvements in relation to benefit to the public health and safety, and other societal and socio-economic considerations, and in relation to the utilization of atomic energy in the public interest." 10CFR50 Appendix I contains further

discussion and specifies as a "guide" the expenditure of \$1000 to achieve a reduction by one man-rem as "reasonable".

The concept of a "hazard index" is of some help in envisioning the magnitude of the problem of protecting people from radioactive substances. The hazard index is the amount of pure water that must be mixed with a given mass of radioactive material to render the mixture acceptable for unrestricted use as drinking water. Figure 4 shows the hazard index of the spent fuel and waste from one ton of fuel. The absolute value is not of great interest except to emphasize the need for care. In section 2 it is stated that 15,000 tons of spent fuel is expected to be withdrawn from reactors in the year 2000. After 1-year's cooling, this would require dilution with 2 million cubic kilometers of water if the mixture or solution is to meet the standard for drinking water. This shows that dilution is an impractical method of disposal.

Of much more significance is the comparison with the radioactivity of the ore required to produce the fuel. The waste and the spent fuel both start with a radioactivity 2000 times above the level of the ore but at all times after 600 years the radioactivity of the reprocessed waste is below that of the ore. At 600 years the spent fuel is about a factor of 5 above the ore and does not reach equality till about 10,000 years. Uranium ore is harmless if buried in its natural site (or at any rate no one is seriously concerned about it at present). If it is exposed by mining without adequate ventilation, it is a serious danger; for example almost all workers in the pitchblend mines of Bohemia in the 16th century died of a mysterious "Bergkrankheit" before they were 40. If the spent fuel or waste could be contained as well as nature contains high grade uranium ore, then the waste disposal problem would be solved for all times over 1000 years (unless of course someone were foolish enough to dig it up).

A.7 LONG-TERM STORAGE OF HIGH-LEVEL WASTE

As shown in Figure A-4, the danger of nuclear waste to living things decreases by several factors of 10 during the first 1000 years after it is produced. After this, decay is very slow and there is no strong reason for choosing any particular time as the limit to the time for which the waste must be isolated. Times of 100,000 to 1 million years are commonly mentioned. The degree of isolation needed and the consequences of failure decrease as time goes on.

The main requirement is to put the waste somewhere where it cannot contaminate air or water; it is also desirable that it should be in a place that is inaccessible to likely human activities. Thus the requirements are for isolation and for inaccessibility. Of these, isolation is the more important since it concerns the prevention of harm to large numbers of people. Inaccessibility is less important since imprudent access or attempts at theft would kill only those few people immediately concerned.

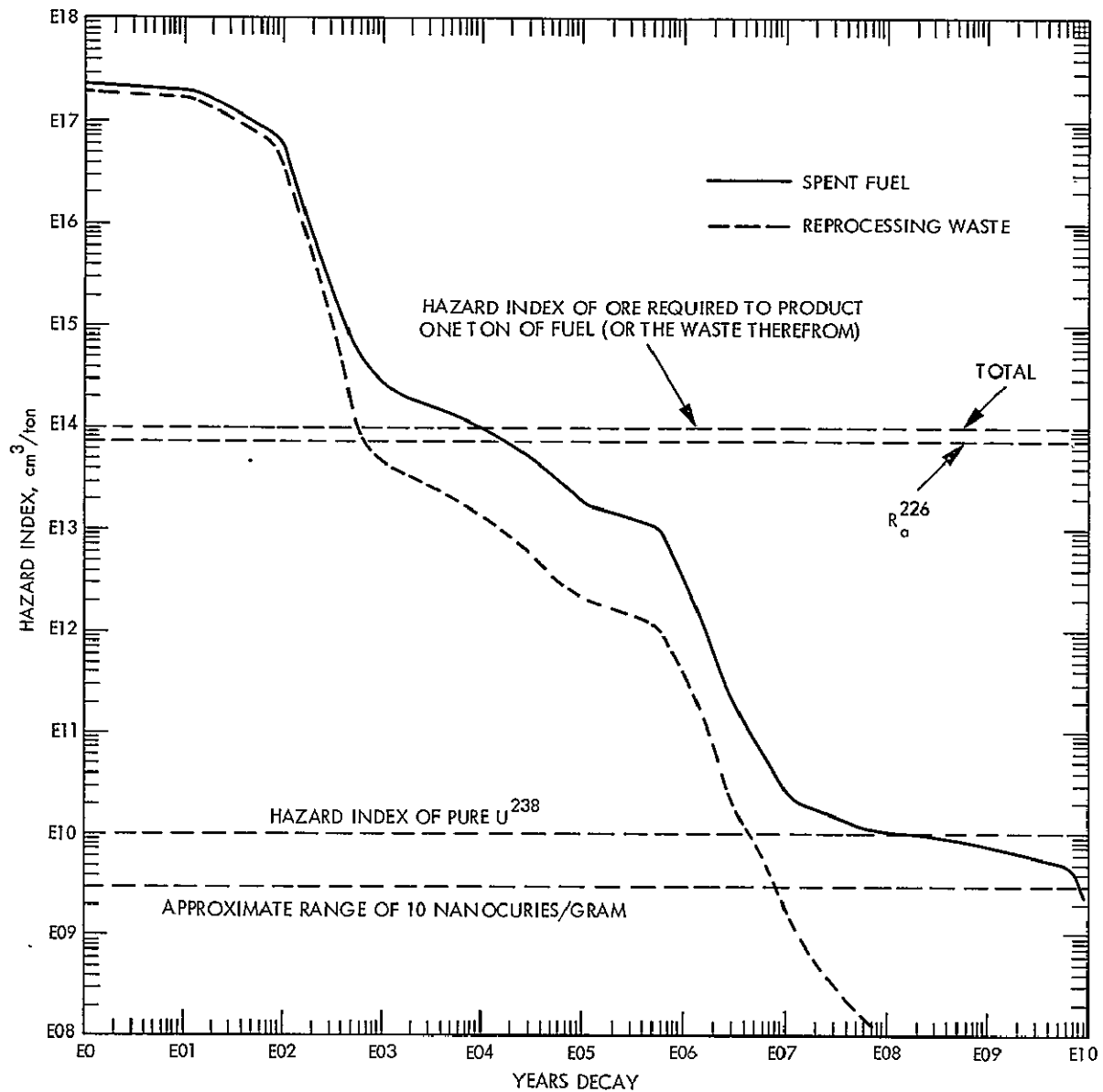


Figure A-4. Hazard Indices for Spent Fuel and Re-processed Waste derived from 1 ton of Fuel. Note: The hazard index for the ore from which the fuel was derived is also shown.

The material to be stored may be either spent fuel in the form of assemblies packed in some kind of container, or the calcined output of a reprocessing plant incorporated in glass. In disposing of spent fuel there is little freedom of choice in the form of the waste. The disposer must take it as he finds it, to chop it up would be to lose much of the advantage of treating the spent fuel itself as waste; extra operations would be needed, subsidiary waste streams would be created, and new paths to the outside world (e.g., the escape of gases) would be created. The only choices are in the form of the container and the means adopted for removing the heat generated in the material.

Most methods for the permanent disposal of spent fuel as waste have a serious long-term disadvantage; they provide posterity with a plutonium mine. After 1000 years the fission products have decayed and, if the spent fuel has been buried, hundreds of tons of plutonium would be available. The balance of risk between burying plutonium and burning it in reactors is by no means clear. Treating spent fuel as waste does not, in itself, remove the "plutonium problem."

The waste from a reprocessing plant contains 200 times less uranium and plutonium than does spent fuel, and it is possible this could be further reduced by a factor of 5 or 10 if this were considered desirable and worth the extra cost. Neptunium could also be removed by the same methods as are employed to remove uranium and plutonium, but the removal of americium and curium is more difficult; methods for this are at present in an early stage of development. If the trans-uranic elements were removed they could be burned in a reactor. Since americium is the element that contributes most to the radioactivity of reprocessed waste for times from 500 to 10,000 years, the matter is of some significance.

If the high-level waste from a reprocessing plant is calcined and incorporated in glass cylinders, there is a good deal of freedom in the choice of the concentration of waste in the glass and in the size of the cylinders. Cylinders of boro-silicate glass 10 feet long and 1 foot in diameter containing 25% of waste are commonly discussed. The choice depends on the temperature that can be tolerated within the glass and thus on the age of the waste. Too high a temperature leads to rapid de-vitrification (that is crystallization) and cracking of the glass, this usually increases the rate at which radioactive material can be leached from the glass. A lower concentration of waste would decrease the temperatures and also decrease the hazard remaining after long times. To take an extreme case, if the waste were diluted in the glass to the point where, after 1000 years, its radioactivity was no greater than that of uranium ore, then it would need no protection other than burial to make it as safe as an ore body. Such dilution is probably too expensive to be practicable, but some concentration of waste in the glass less than the maximum technically possible may be a worthwhile compromise. Another possibility is deliberately to crystallize the glass-waste mix and sinter the crystals into a ceramic. Considerable investigation would be needed to determine how to produce crystals that would incorporate the principal radioactive elements into the crystal lattice.

When it has been decided what it is that is to be stored, it is necessary to choose a place and a method of storage. Many places and methods have been suggested and some have been worked out in considerable detail. The principal ones are:

- (1) Deep Geological disposal, that is, deep burial on land. This is the method proposed by ERDA for the first repository which is planned to be in operation in 1985.
- (2) Geological disposal beneath the sea floor.
- (3) Disposal in ice sheets.
- (4) Engineered surface, or near surface, storage.
- (5) Extra-terrestrial disposal in earth orbit, solar orbit, on the moon or by ejection out of the solar system or into the sun.

These concepts are each considered below, most attention being paid to the ones that are nearest to implementation.

(1) Geologic storage on land. Oil and gas, as found in nature, have commonly been stored where they are found for 10 to 100 million years. An oil field consists essentially of a porous rock containing oil and gas kept from escaping by an impermeable layer (salt, clay or shale are typical materials in the impermeable layer) and a geometry, such as an arch (an "anticline"), that allows the oil and gas to be trapped in a limited area. The success of nature's system of storage suggests that geological storage of nuclear waste for long periods may be possible.

Storage of waste in liquid form is not seriously considered, probably because a liquid is inherently more mobile than a solid and large and sudden escapes through cracks could conceivably occur. Storage of solids in an excavated cavity is the method usually suggested. One or more shafts would be excavated and a series of horizontal tunnels bored outwards from the shafts; the storage would be in vaults, leading off from the tunnels. In order to provide support to the roof and shielding from γ -rays, the walls between the vaults would be of a thickness about the same as the width of the vaults. The depository would initially be operated in such a way that the waste could be recovered. Later, when confidence had been gained by some years of experience, the excavation would be back filled and the waste regarded as permanently entombed. Extensive monitoring would be carried out in the early stages to detect unexpected leaks. The spacing of individual containers would be sufficient to avoid criticality following an extremely severe accident, such as complete flooding of the vaults.

Such a disposal scheme provides a series of barriers to the spread of radioactivity. The main ones are:

- (1) The glass

(2) The canister

(3) The surrounding rock (including hold up by absorption)

If the vault remains dry and the waste has cooled for 10 or 20 years before emplacement, the glass, and the canister should provide useful isolation during the time when the radioactivity is at its greatest intensity. If the vault is flooded, which is unlikely but not impossible, the canister and the glass could not be relied on indefinitely to prevent dispersion of the radioactive materials.

Thus, in the worst case, the security of the isolation depends not only on the packaging but also on the slowness of the migration of the radioactive material through the overlying rock. Diffusion of water through intact rock is slow. The diffusion of radioactivity is even slower, particularly through a fine grained sediment, since it is held back by absorption on the surfaces of the mineral grains. Experimental and theoretical studies suggest (e.g. Reference A-7.) that many geological materials would be satisfactory from this point of view. The slowing of the spread of radioactivity is very great, factors of 100 to 1,000,000 being found.

Unfortunately, diffusion along grain boundaries is not the only process for the movement of solutions in rocks. It is also possible for fluids to move by bulk flow through cracks and to emerge as springs. This is perhaps the most dangerous pathway by which buried waste could reach the surface. The first requirement for the rock surrounding a waste repository is that it should not be cracked, it is also necessary that it should be impervious (for example most sandstones would be unsuitable). In general, hard rocks which have solidified from the molten state are liable to cracking, and a very careful study would be necessary to ensure that the rock at a particular site was not cracked. This has been recognized in Sweden where it is proposed that waste should be stored in a granite repository and surrounded on all sides by 3 m of clay.

In limestone there is a special danger that cracks can be widened by solution and form caves and underground rivers. No doubt there are bodies of granite, basalt, or limestone that are free from cracks and suitable for the excavation of a repository; the difficulty is to demonstrate the soundness of the rock before the repository is excavated and to be sure that cracks will not develop as a result of excavation.

The main attraction of rock salt (see Figure A-5) is that it is an impervious medium in which cracks can be healed by flow of the slightly plastic material. There is extensive experience in mining salt and in containing fluids in large cavities. Natural traps for oil and gas have also been studied in great detail by the oil industry. The main new feature introduced by the storage of waste is the generation of heat in the vault. This may have local effects in breaking up the salt and possibly more widespread effects which may be more important. As the heat spreads out from the vault it is retained in the overlying rocks for many thousands of years. The heating will cause the rocks to expand and will produce stresses. In a plastic medium such as salt these will be taken up by flow, and no serious consequences are likely to follow.

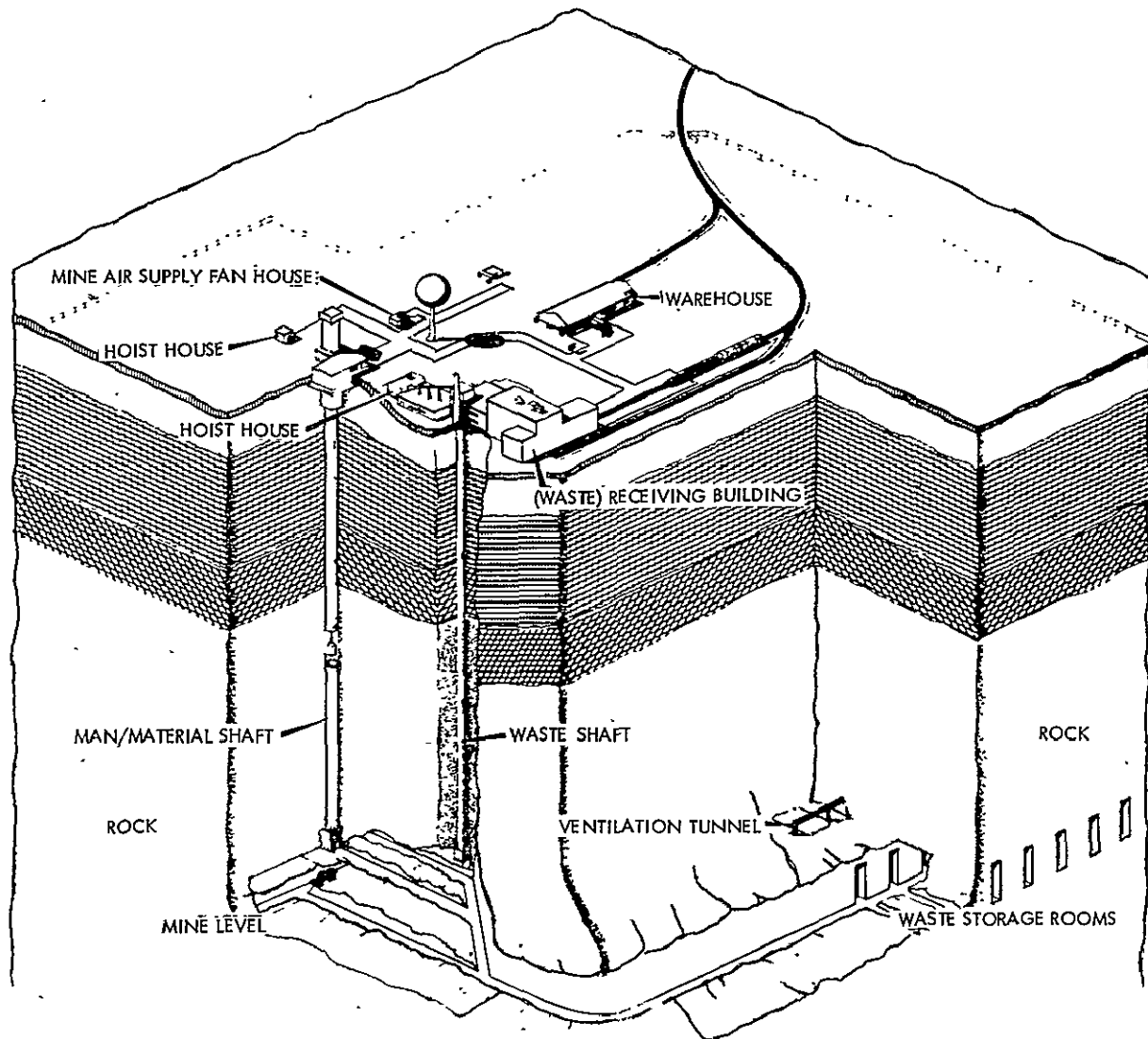


Figure A-5. Underground Storage in Bedded Salt

In some districts where salt occurs, there are salt springs. These are usually formed by the flow of surface water through porous beds overlying the salt; the water cannot penetrate the salt and therefore flows along its surface, dissolves as much salt as it can hold, and emerges again, usually at a lower level in a valley. Clearly, it is desirable to choose a salt bed overlain by impervious material such as shale or clay. The effectiveness of this is shown by the survival of salt beds beneath the floor of the Red Sea, a layer of fine grained sediment has protected the salt for over 10 million years. It is also desirable that the depository be below the level of the bottom of all nearby valleys. An overlying layer of clay or shale also has the advantage that, as described above, it will greatly retard the movement of the radioactive materials if they escape from the glass, the canister and the salt.

Storage in salt has been explored in the U.S. without any irremediable difficulty being found. There are sites where a repository can be expected to remain dry for an indefinite period and where flooding, if it occurred, is not expected to cause disastrous results. The literature on this subject includes a discussion by ERDA (Reference A-1, Volume 4), preliminary performance criteria and fault trees are discussed in Reference A-8.

(2) Geological Storage Beneath the Sea Floor (Reference 9). The bottom of the deep sea is the most remote and least frequented part of the earth's surface. In depths exceeding 5000 m it is usually floored by "red clay" which is a fine-grained mud or clay composed of insoluble grains left from the solution of shells or skeletons of calcareous animals, volcanic ash, and chemically-precipitated material, mostly iron and manganese oxides. It has been suggested that waste could be emplaced in this mud by drilling a hole in a place where the sediment is several 100 m in thickness. The techniques of drilling in deep water are well-developed, and there is no difficulty in principle in emplacing the waste and filling the hole afterwards. It is difficult to predict what proportion of canisters would stick at less than the intended depth.

The choice of site would be important. The surface of the earth is divided into plates which move at a speed of 1 to 10 cm/year away from the mid-ocean ridges, where they are created, towards the ocean trenches, where they plunge downwards into the interior of the earth. It has been suggested that the waste might be deposited near ocean trenches. This seems imprudent since the sediment on the downgoing plate is often scraped off and piled up on the edge of the island arc or continent beneath which the plate is sinking. In addition the speed of the plate (less than 1km in 10,000 years) is too slow to be useful.

A much more favorable site is provided by the center of a plate. Here the site is away from the earthquakes of the ridge, which are associated with its creation and is also far from the earthquakes associated with its destruction. The middle of a plate is not entirely free from earthquakes, but they are as rare as they are anywhere on earth. The main doubts about the disposal of waste at mid-plate sites is the effect of the heat generated by the waste on the stability of the clay in which it is emplaced. This is a matter which requires further investigation. Work is at present proceeding on a rather small scale to establish

the scientific principles on which the practicability of sea bed storage can be judged (Reference A-9).

(3) Storage in Ice Sheets. It would be easy to emplace canisters of waste deep in an ice sheet; in fact, a canister placed on the surface would melt its way to the bottom. If the ice were moving, the canister would be torn to pieces either by the shearing of the ice or by being dragged along the rocky bottom of the ice sheet. What would happen then depends on whether the ice is melted at the bottom, (as it is in the glaciers of temperate climates) and on whether the water can escape. Doubtless these questions about the movement of ice and water and the possibility of the waste re-appearing at the edge of the ice sheet could be resolved; however, questions of climatic stability on the required time scale cannot be answered in the present state of Climatology. The two largest ice sheets are in Greenland and Antarctica. Greenland belongs to Denmark and there is a treaty forbidding the use of Antarctica. There seems no reason to regard disposal of waste in ice sheets as a method likely to be used in the foreseeable future.

(4) Engineered storage. Surface structures have been successfully used for temporary storage of spent fuel and less successfully for processed waste from military programs. There appears to be no technical reason why it should not be successful for, say, 1000 years, but for a much longer term, many doubts arise. The doubts are not only technical, such as changes of climate and flooding of sites, but also political. Any surface structure is subject to destruction by human interference, and it seems, on the whole, imprudent to embark on such a scheme for the storage of very active commercial waste.

(5) Extra-terrestrial disposal. Extra-terrestrial disposal is attractive in that it permanently removes objectionable material from the earth. The objections to this disposal option are:

- (a) The possibility of accidents in which a consignment crashes on land or burns in the atmosphere; to avoid this and give safe re-entry and landing would greatly increase the weight to be transported.
- (b) The possibility of contamination of the stratosphere with the products of combustion of the fuel needed to get the consignment off the earth.
- (c) The cost which is likely to significantly exceed that of a geological repository.

The whole question of space disposal is being investigated by NASA who are considering many possible options. Until these efforts have reached their conclusions, it would be premature to discuss the matter further.

It appears clear that the only permanent storage facility that is likely to be available during the 1980s is the one being designed by ERDA to be constructed in bedded salt. This has received far more

attention than any other method. It is ERDA's intention to operate it in a way that will allow the waste to be recovered if irremediable faults are found before the disposal site is converted from a retrievable to an irretrievable mode. The main problem is to be sure that a particular site is similar to the ideal site for which the plans are made and that there are no cracks that allow water to circulate in the salt and which may allow the water to get into the workings.

Other materials are also being investigated including basalt, granite, shale and clay. Since Hanford is underlain by basalt it would be most economical if its waste could be safely deposited in basalt. The main problem in igneous rocks is to demonstrate the absence of cracks. A suitably fine grained shale might prove an excellent choice, but no extensive study has been made.

A.8 OTHER FUEL CYCLES

This discussion has been directed to the high-level waste from a PWR fueled with slightly enriched uranium. There are many other types of reactors in use and still more are possible. The problems posed by the waste are very similar for all of them. All produce about the same amount of fission products for a given amount of power. The transuranic elements differ more than do the fission products between different reactors. All give the same isotopes but the proportions of them depend on the fuel and on the neutron spectrum. These differences do not qualitatively affect the problems of storage, certainly not to the extent that the decision to reprocess or not to reprocess does; it is therefore unnecessary to go into the details of waste from other types of reactors in a background paper such as this.

A.9 REFERENCES

- A-1 Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle, Volume 1., 1976. ERDA-76-43
- A-2 Blomeke, J. O., Kee, C. W.; and Nichols, J. P., Projections of Radioactive Wastes to be Generated by the U.S. Nuclear Power Industry. 1974. ORNL-TM-3965.
- A-3 Kee, C. W.; Croff, A. G.; and Blomeke, J. O., Updated Projections of Radioactive Wastes to be Generated by the U.S. Nuclear Power Industry. 1976. ORNL/TM-5427
- A-4 LWR Spent Fuel Disposition Capabilities 1976-1985. 1976. ERDA-76-25.
- A-5 The Effects on Populations of Exposure to Low Levels of Ionising Radiation. 1972. National Academy of Sciences.
- A-6 Lieberman, J. A., Rodger, W. A., and Baranowski, F. P., Testimony to ERCDCC, 1977
- A-7 Lester, D. H., Janse, G., and Burkholder, H. C., Migration of Radionuclide Chains Through an Absorbing Medium. Amer. Inst. Chem. Eng., Symposium Series No. 152 Absorption and Ion Exchange 71, 202, 1975
- A-8 Determination of Performance Criteria for High-Level Solidified Nuclear Waste. Preliminary Report. Lawrence Livermore Laboratory NUREG-1002.
- A-9 Oceanus, April 1977—

APPENDIX B

GLOSSARY

The following list of terms abbreviations, and acronyms frequently used in discussions of nuclear waste management was prepared for the staff of the California Energy Resources Conservation and Development Commission. The selection of terms was based on experience with the relevant literature.

Sources for the terms include:

- (1) Nuclear Terms, A Brief Glossary, U.S. Atomic Energy Commission, available from U.S. ERDA, Division of Technical Information Extension, Oak Ridge, Tennessee.
- (2) "Glossary", Oceanus, William H. MacLeish, Editor, Volume 20 Number 1, p. 64, Woods Hole Oceanographic Institution, Woods Hole, Massachusetts, 02543, Winter 1977.
- (3) Alternatives for Managing Wastes from Reactors and Post-fission Operations in the LWR Fuel Cycle, ERDA-76-43, Volume 5, Appendix A, available from National Technical Information Service, Springfield Virginia, 22151, May 1976.
- (4) Willrich, M., Radioactive Waste Management and Regulation, Report to the Energy Research and Development Administration from the M.I.T. Energy Laboratory (draft, MIT-EL-76-011, Sept. 1, 1976.

Quantitative prefixes and their symbols are listed on the last page of this Appendix.

Absorbed Dose	When Ionizing Radiation passes through matter, some of its energy is imparted to the matter. The amount absorbed per unit mass of irradiated material is called the absorbed dose, and is measured in rems and rads. (See <u>threshold dose</u> .)
Absorber	Any material that absorbs or diminishes the intensity of ionizing radiation. Neutron absorbers, like boron, hafnium, and cadmium, are used in control rods for reactors. Concrete and steel absorb gamma rays and neutrons in reactor shields. A thin sheet of paper or metal will absorb or attenuate low energy alpha particles and all, except for the most energetic of the beta particles. (Compare <u>moderator</u> ; see <u>poison</u> .)
Absorption	The process by which the number of particles or photons entering a body of matter is reduced by interaction of the particles or radiation with the matter; similarly, the reduction of the energy of a particle while traversing a body of matter. This term is sometimes erroneously used for capture. (Compare <u>capture</u> ; see <u>stopping power</u> .)
Actinide Series	The series of elements beginning with actinium element No. 89 and continuing through lawrencium element No. 103, which together occupy one position in the <u>Periodic Table</u> . The series includes uranium element No. 92 and all the transuranic elements. The group is also referred to as the "Actinides". (Compare <u>lanthanide series</u> , <u>transuranic elements</u> .)
Activation	The process of making a material radioactive by bombardment with neutrons, protons, or other nuclear particles. Also called radioactivation.
ACVC	<u>Air Cooled Vault Concept</u> . A concept developed as an option for RSSF and for SURFF. Suggested in WASH 1539.
AEC	U.S. <u>Atomic Energy Commission</u> . In 1975, the Atomic Energy Commission was divided into two separate agencies. The regulatory portion became the Nuclear Regulatory Commission, and the reactor development portion became part of the Energy Research and Development Administration.

Aftercooling	The cooling of a reactor after it has been shut down.
Afterheat	The heat produced by the continuing decay of radioactive atoms in a reactor after fission has stopped. Most of the afterheat is due to the radioactive decay of <u>fission products</u> .
AGNS	<u>Allied-General Nuclear Services</u> .
Alpha Particle	A positively charged particle emitted by certain radioactive materials. It is made up of two neutrons and two protons bound together. It is the least penetrating of the three common types of radiation (alpha, beta, gamma) emitted by radioactive material, and is usually stopped by a sheet of paper.
Aqueous Raffinate	Liquid left in the solvent extraction system, from which the uranium has been extracted by contact with an immiscible (two liquids that do not mix and form more than one phase when brought together) organic solvent. The solvent is tributylphosphate in hexane (TBP).
Atom	A particle of matter indivisible by chemical means. It is the fundamental building block of the chemical elements. The elements, such as iron, lead, and sulfur, differ from each other because they contain different kinds of atoms. There are about 6×10^{21} atoms in an ordinary drop of water. According to present day theory, an atom contains a dense inner core (the nucleus) and a much less dense outer domain consisting of electrons in motion around the nucleus. Atoms are electrically neutral. (Compare <u>element</u> , <u>ion</u> , <u>molecule</u> ; see <u>matter</u> .)
Atomic Explosion	An explosion in which energy is produced by nuclear <u>fission</u> or <u>fusion</u> . (Compare <u>device</u> , <u>nuclear</u> .)
Atomic Weight	The mass of an atom relative to other atoms. The present day basis of the scale of atomic weights is carbon; the commonest isotope of this element has arbitrarily been assigned an atomic weight of 12. The unit of the scale is 1/12 the weight of the carbon 12 atom, or roughly the mass of one proton or one neutron. The atomic weight of any element is approximately equal to the total number of protons and neutrons in its nucleus.

Back End of the Fuel Cycle	Includes reactors, spent fuel storage, fuel reprocessing, mixed-oxide fuel fabrication, transportation, and waste management.
Background Radiation	The radiation in man's natural environment, including cosmic rays and radiation from the naturally radioactive elements, both outside and inside the bodies of men and animals. It is also called <u>natural radiation</u> . The term may also mean radiation that is unrelated to a specific experiment.
Barrier Shield	A wall or enclosure shielding the operator from an area where radioactive material is being used or processed by remote control equipment.
Beta Particle	An elementary particle emitted from a nucleus during radioactive decay, with a single electrical charge and a rest mass equal to $1/1837$ that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is called a positron. Beta radiation may cause skin burns, and beta emitters are harmful if they enter the body. Beta particles are easily stopped by a thin sheet of metal.
Benchmark	A standard by which others may be measured or evaluated.
BeV	Symbol for billion (or 10^9) electron volts.
Biological Dose	The radiation dose absorbed in biological material. Measured in rems. (See <u>absorbed dose</u> .)
Biological Half-life	The time required for a biological system, such as a man or an animal, to eliminate, by natural processes, half the amount of a substance (such as a radioactive material) that has entered it. (Compare <u>half-life</u> ; see <u>half-life, effective</u> .)
Biological Shield	A mass of absorbing material placed around a reactor or radioactive source to reduce the radiation to a level that is safe for human beings. (See <u>absorber</u> , <u>shield</u> , <u>thermal shield</u> .)
Biosphere	That part of the earth (upwards at least to a height of 10,000 m, and downwards to the depths of the ocean, and a few hundred meters below the ground surface) and the atmosphere and hydrosphere surrounding it, which is able to support life.

Blanket	A layer of fertile material, such as uranium 238 or thorium-232, placed around the fissionable material in a reactor. (See <u>fertile material</u> , seed core.)
BNFP	Barnwell Nuclear Fuel Plant -- a reprocessing plant in South Carolina which has not been licensed for operation.
BNWL	Battelle Northwest Laboratories.
Body Burden	The amount of radioactive material present in the body of a man or an animal.
Boiling Water Reactor (BWR)	A reactor in which water, used as both coolant and moderator, is allowed to boil in the core. The resulting steam can be used directly to drive a turbine.
Bone Seeker	A <u>radioisotope</u> that tends to accumulate in the bones when it is introduced into the body. An example is strontium-90, which behaves chemically like calcium.
Breeder Reactor	A reactor that produces fissionable fuel as well as consuming it, especially one that creates more than it consumes. A new <u>fissionable material</u> is created by capture in fertile materials of neutrons from fission. The process by which this occurs is known as <u>breeding</u> . (Compare <u>converter reactor</u> ; <u>LMFBR</u> see <u>fertile material</u> .)
Breeding Ratio	The ratio of the number of fissionable atoms produced in a breeder reactor to the number of fissionable atoms consumed in the reactor. <u>Breeding gain</u> is the breeding ratio minus one. (Compare <u>conversion ratio</u> .)
Burial Grounds	Areas designated for storage of containers of treated radioactive wastes by burial in geologic media.
Burnable Poison	A neutron absorber (or poison), such as boron, which, when purposely incorporated in the fuel or fuel cladding of a nuclear reactor, gradually "burns up" (is changed into nonabsorbing material) under neutron irradiation. This process compensates for the loss of reactivity that occurs as fuel is consumed and fission-product poisons accumulate, and keeps the overall characteristics of the reactor nearly constant during its use. (See <u>poison</u> , <u>reactivity</u> .)
Burner Reactor	A converter reactor

Burnup	A measure of reactor fuel consumption. It can be expressed as (a) the percentage of fuel atoms that have undergone fission, or (b) the amount of energy produced per unit weight of fuel in the reactor.
By-Product Material	Any radioactive material (except source material or fissionable material) obtained during the production or use of source material or fissionable material. It includes fission products and many other radioisotopes produced in nuclear reactors. (Compare <u>fissionable material, source material</u> .)
BWR	<u>Boiling Water Reactor</u>
Capture	A process in which an atomic or nuclear system acquires an additional particle; for example, the capture of electrons by positive ions, or capture of electrons or neutrons by nuclei.
Calcine	To calcine; generally, to heat (or roast) ores or ore concentrates so as to remove its water, CO_2 , or sulfur content. In nuclear technology, to roast pre-dried ammonium diuranate (ADU) in a reducing atmosphere (H_2 and N_2) so as to obtain UO_2 in a powdery form. In the reprocessing of high-level liquid wastes, to calcine is to solidify, through roasting, waste material and convert it to a high temperature stable oxide.
Calcines	The UO_2 powder which comes from roasting pre-dried ammonium diuranate (ADU) in a reducing atmosphere (H_2 and N_2). This pertains to the fuel fabrication process (cf. LWR Fuel Cycle). In the solidification reprocessing of high-level liquid wastes, the end product in the form of powdery or granular material or a porous cake, depending on the particular process.
Carrier	A stable isotope, or a normal element to which radioactive atoms of the same element can be added to obtain a quantity of radioactive mixture sufficient for handling, or to produce a radioactive mixture that will undergo the same chemical or biological reaction as the stable isotope. A substance in weighable amount which, when associated with a trace of another substance, will carry the trace through a chemical, physical or biological process.
Cascade	A connected arrangement of units of equipment for separation of isotopes. A single device or process usually can produce only a small amount of isotopic separation, but if a number of these are connected

	together the effect can be multiplied and a significant amount of separation achieved. An example is a cascade of barriers for the gaseous diffusion process. (See <u>gaseous diffusion, isotope separation.</u>)
CFR	Code of Federal Regulations is a codification of the general and permanent rules published in the Federal Register by the Executive departments and agencies of the Federal Government.
Chain Reaction	A reaction that stimulates its own repetition. In a fission chain reaction a fissionable nucleus absorbs a neutron and fissions, releasing additional neutrons. These in turn can be absorbed by other fissionable nuclei, releasing still more neutrons. A fission chain reaction is self-sustaining when the number of neutrons released in a given time equals or exceeds the number of neutrons lost by absorption in non-fissioning material or by escape from the system.
Cladding	The outer jacket of nuclear fuel elements. It prevents corrosion of the fuel and the release of fission products into the coolant. Aluminum or its alloys, stainless steel and Zirconium alloys are common cladding materials.
Cladding Waste	Spent fuel rods after removal from a reactor are broken up and fissionable fuel is chemically leached out. The remaining residue, principally radioactive cladding material, insoluble nuclear fuel, fission products, and transuranium nuclides, is called cladding waste.
Closed-Cycle Reactor System	A reactor design in which the primary heat of fission is transferred outside the reactor core to do useful work by means of a coolant circulating in a completely closed system that includes a heat exchanger. Compare <u>direct-cycle reactor system, indirect-cycle reactor system, open-cycle reactor system.</u>)
Coffin	A heavily shielded shipping cask for spent (used) fuel elements. Some coffins weigh as much as 75 tons.
Containment	The provision of a gastight low pressure shell or other enclosure around a reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident.
Containment Vessel	A gas tight low pressure shell or other enclosure around a reactor. (Compare <u>pressure vessel</u> ; see <u>containment.</u>)

Contamination	(See <u>radioactive contamination</u> .)
Control Rod	A rod, plate, or tube containing a material that readily absorbs neutrons (hafnium, boron, etc.), used to control the power of a nuclear reactor. By absorbing neutrons, a control rod prevents the neutrons from causing further fission. (See absorber).
Conversion Ratio	The ratio of the number of atoms of new fissionable material produced in a converter reactor to the original number of atoms of fissionable fuel consumed. (Compare <u>breeding ratio</u> .)
Converter Reactor	A reactor that produces some fissionable material, but less than it consumes. In some usages, a reactor that produces a fissionable material different from the fuel burned, regardless of the ratio. In both usages the process is known as conversion. (Compare <u>breeder reactor</u> .)
Coolant	A substance circulated through a nuclear reactor to remove or transfer heat. Common coolants are water, air, carbon dioxide, liquid sodium, and sodium potassium alloy (NaK).
Cooling Pond	An open pond in which water, heated through use in an industrial process is allowed to cool through evaporation before re-use.
Core	The central portion of a nuclear reactor containing the fuel elements and usually the moderator, but not the reflector.
Critical	Capable of sustaining a chain reaction. (See <u>criticality</u> .)
Critical Assembly	An assembly of sufficient fissionable material and moderator to sustain a fission chain reaction at a very low power level. This permits study of the behavior of the components of the assembly for various fissionable materials in different geometrical arrangements. (Compare <u>nuclear reactor</u> .)
Critical Facility	A facility where a critical mass of fissionable material or sub critical mass of fissionable material is used in experiments or stored.
Critical Mass	The smallest mass of fissionable material that will support a self-sustaining chain reaction under stated conditions.

Criticality	The state of a nuclear reactor when it is sustaining a chain reaction. (See <u>dry criticality, multiplication.</u>)
Curie	The basic unit to describe the intensity of radioactivity in a sample of material. The Curie is equal to 37 billion disintegrations per second (3.7×10^{10} dis/sec), which is approximately the rate of decay of 1 gram of radium. A curie is also a quantity of any nuclide having 1 curie of radioactivity. Named for Marie and Pierre Curie who discovered radium in 1898. (Compare <u>rem, roentgen.</u>)
Daughter	A nuclide formed by the radioactive decay of another nuclide, which in this context is called the parent.
Decay Chain	A radioactive series.
Decay Heat	The heat produced by the decay of radioactive nuclides. (See <u>afterheat; decay, radioactive; SNAP.</u>)
Decay, Radioactive	The spontaneous transformation of one nuclide into a different nuclide or into a different energy state of the same nuclide. The process results in a decrease, with time, of the number of the original radioactive atoms in a sample. It involves the emission from the nucleus of alpha particles, beta particles (or electrons), or gamma rays; or the nuclear capture or ejection of orbital electrons. Also called radioactive disintegration. (See <u>half-life, nuclear reaction, radioactive series.</u>)
Decommissioning	Preparation of worn out or obsolete nuclear facilities for retirement. Decommissioning operations remove facilities such as reactors, reactor containments, reprocessing plants, and burial grounds from service and reduce or stabilize radioactive contamination. Concepts include dismantling, entombment, and mothballing which evolve: ● Demolition and restoration to original conditions requiring no control ● Partial demolition and fixation of residues ● Minimal demolition followed by isolation and control of residues
Decontamination	The removal of radioactive contaminants from surfaces or equipment, as by cleaning and washing with chemicals. (See <u>radioactive contamination.</u>)

Deep Continental Geologic Formations	Geologic media beneath the continents and isolated from biologic species and phenomena. Distinguished from ice sheets and sea floor geologic media.
Depleted Fuel	(See <u>depleted uranium, spent fuel.</u>)
Depleted Uranium	Uranium having a smaller percentage of uranium-235 than the 0.7% found in natural uranium. It is obtained from the spent (used) fuel elements or as by-product tails, or residues, of uranium isotope separation. (Compare natural uranium, spent fuel.)
Deuterium	An isotope of hydrogen whose nucleus contains one neutron and one proton and is therefore about twice as heavy as the nucleus of normal hydrogen which has only a single proton. Deuterium is often referred to as heavy hydrogen; it occurs in nature as 1 atom to 6500 atoms of normal hydrogen. It is nonradioactive. (See <u>heavy water, heavy hydrogen.</u>)
Difussion Plant	(See <u>gaseous diffusion.</u>)
Direct-Cycle- Reactor System	A nuclear power plant system in which the coolant or heat transfer fluid circulates first through the reactor and then directly to a turbine. (Compare <u>indirect-cycle reactor system.</u>)
Disintegration, Radioactive Disposal	Equivalent to radioactive decay. Operations designed to eliminate wastes from existence on earth or to permanently isolate them from mankind and his environs with no expectation of retrieval after emplacement. Isolation concepts include ● placement in subsurface geologic formations using technologies that offer no practical method for recovery ● emplacement into or beneath sea floors ● dispersion Elimination concepts include extraterrestrial disposal and transmutation.
Distribution Factor	A term used to express the modification of the effect of radiation in a biological system attributable to the nonuniform distribution of an internally deposited isotope, such as radium, being concentrated in bones. (See <u>absorbed dose, dose equivalent, quality factor.</u>)

Dollar	A unit of reactivity. One dollar is the maximum amount of reactivity in a reactor due to delayed neutrons alone.
Dose	(See absorbed dose, biological dose, maximum permissible dose, threshold dose.)
Dose Equivalent	A term used to express the amount of effective radiation when modifying factors have been considered. The product of absorbed dose multiplied by a quality factor multiplied by a distribution factor. It is expressed numerically in rems.
Dose Rate	The radiation dose delivered per unit time and measured, for instance, in rems per hour. (See absorbed dose, rem.)
Doubling Time	The time required for a breeder reactor to produce as much fissionable material as the amount usually contained in its core plus the amount tied up in its fuel cycle (fabrication, reprocessing, etc.). It is estimated as 10 to 20 years in typical reactors. (See breeder reactor, fuel cycle.)
Dry Criticality	Reactor criticality achieved without a coolant. (Compare wet criticality; see criticality.)
Dual-Cycle Reactor System	A reactor-turbine system in which part of the steam fed to the turbine is generated directly in the reactor and part in a separate heat exchanger. A combination of direct cycle and indirect-cycle reactor systems.
Dual Purpose Reactor	A reactor designed to achieve two purposes, for example, to produce both electricity and new fissionable material.
Effective Half-life	(See half-life, effective.)
Effective Multiplication factor (or constant)	(See <u>multiplication factor</u> .)
Electromagnetic Radiation	Radiation consisting of associated and interacting electric and magnetic waves that travel at the speed of light. Examples: light, radio waves, <u>gamma rays</u> , <u>X-rays</u> . All can be transmitted through a vacuum. (Compare <u>ionizing radiation</u> ; see <u>quantum</u> .)

Element	One of the over 100 known chemical substances that cannot be divided into simpler substances by chemical means. A substance whose atoms all have the same <u>atomic number</u> . Examples: hydrogen, lead, uranium. (Not to be confused with <u>fuel element</u> .) (See atom, <u>matter</u> , <u>nuclide</u> .)
Encapsulate	To encapsulate: To hermetically seal the container that has in it one or several spent fuel assemblies and the "filler material" which enhances heat conductivity.
Energy Research and Development Administration (ERDA)	In 1975, the Atomic Energy Commission was divided into two new agencies. The regulatory portion became the Nuclear Regulatory Commission and the reactor development portion became part of the <u>Energy Research and Development Administration</u> .
Engineered Storage	The storage of radioactive wastes, usually within suitable-sealed containers, in any of a variety of structures especially designed to protect them from water and weather, and to help keep them from leakage to the biosphere by accident or sabotage. They may also provide for extracting heat of radioactive decay from the waste.
Enriched Material	Material in which the percentage of a given isotope present in a material has been artificially increased, so that it is higher than the percentage of that isotope naturally found in the material. Enriched uranium contains more of the fissionable isotope uranium-235 than the naturally occurring percentage (0.7%). (See <u>isotopic enrichment</u> .)
Enrichment	<u>Isotopic enrichment</u> .
EPA	<u>Environmental Protection Agency</u> . EPA's statutory authorities for radiation protection are primarily concerned with the promulgation of environmental standards and guidelines. EPA also has limited authority for effluent regulation and enforcement activities in some areas; however this responsibility is mainly under the purview of NRC
Excess Reactivity	More reactivity than that needed to achieve criticality. Excess reactivity is built into a reactor (by using extra fuel) in order to fuel <u>burnup</u> and the accumulation of <u>fission-poisons</u> during operation. (See <u>criticality</u> , <u>reactivity</u> .)

Exclusion Area	An area immediately surrounding a nuclear reactor where human habitation is prohibited to assure safety in the event of accident. (See <u>low population zone</u> .)
Excursion	A sudden, very rapid rise in the power level of a reactor caused by <u>supercriticality</u> . Excursions are usually quickly suppressed by the negative temperature coefficient of the reactor and/or by automatic control rods.
Experimental Reactor	A reactor to test the design of new reactors. (Compare <u>research reactor</u> , <u>test reactor</u> .)
External Radiation	Radiation from a source outside the body.
Fast Breeder Reactor	A reactor that operates with fast neutrons and produces more fissionable material than it consumes. (See <u>breeder reactor</u> , <u>fast neutron</u> , <u>fast reactor</u> .)
Fast Neutron	A neutron with energy greater than approximately 100,000 electron volts. (Compare <u>intermediate neutron</u> , thermal neutron.)
Fast Reactor	A reactor in which the fission chain reaction is sustained primarily by fast neutrons rather than by thermal or intermediate neutrons. Fast reactors contain little or no moderator to slow down the neutrons from the speeds at which they are ejected from fissioning nuclei. (Compare intermediate reactor, thermal reactor.)
Federal Repository	Federally operated disposal or storage facility for high-level and transuranic contaminated wastes.
Feed Materials	Refined uranium or thorium metal or their pure compounds in a form suitable for use in nuclear reactor fuel elements or as feed for uranium enrichment processes. (See enriched material.)
Fertile Material	A material, not itself fissionable by thermal neutrons, which can be converted into a fissile material by irradiation in a reactor. There are two basic fertile materials, uranium-238 and thorium-232. When these fertile materials capture neutrons.
Final Storage	Storage operations for which a) no subsequent waste treatment or transportation operations are anticipated and b) conversion to disposal (i.e., termination of monitoring and human control) is considered possible.

The current concept for final storage is emplacement of wastes in geologic formations. The geologic formations may be either on the earth's surface and augmented by engineering technology (e.g., burial grounds), or subsurface and augmented with engineering technology. Fissile material, while sometimes used as a synonym for fissionable material, this term has also acquired a more restricted meaning, namely, any material fissionable by neutrons of all energies, including (and especially) thermal (slow) neutrons as well as fast neutrons; for example, uranium-235 and plutonium-239. (See fissionable material.)

Fission

The splitting of a heavy nucleus into two approximately equal parts (which are nuclei of lighter elements), accompanied by the release of a relatively large amount of energy and generally one or more neutrons. Fission can occur spontaneously, but it usually is caused by nuclear absorption of neutrons, gamma rays, or other particles. (Compare fusion; see chain reaction, nuclear reaction.)

Fission Fragments

The two nuclei which are formed by the fission of a nucleus. Also referred to as primary fission products. They are of medium atomic weight, and are radioactive. (See fission products.)

Fission-product Poisoning

The absorption or capture of neutrons by fission products in a reactor, decreasing its reactivity. (See poison.)

Fission Products

The nuclei (fission fragments) formed by the fission of heavy elements, elements, plus the nuclides formed by the fission fragments' radioactive decay. (Compare fission fragments; see decay, radioactive.)

Fission Yield

The amount of energy released by fission in a thermonuclear (fusion) explosion as distinct from that released by fusion. Also the amount (percentage) of a given nuclide produced by fission. (Compare yield; see thermonuclear reaction, TNT equivalent.)

Fissionable Material

Commonly used as a synonym for fissile material. The meaning of this term also has been extended to include material that can be fissioned by fast neutrons only, such as uranium-238. Used in reactor operations to mean fuel. (Compare fertile material, fissile material.)

Fluidized Bed Calciner	Fluidized bed calciner, used to convert high level liquid radioactive waste (HLLW) to a dry solid. The HLLW is atomized into a fluidized bed, heated by inbed combustion (kerosene jet burning with oxygen input), the bed temperature being 500°C - 600°C. Evaporation occurs at the surface of the particles and results in granular bed material and powdered calcine.
Fluid Fuel Reactor	A type of reactor (for example, a <u>fused-salt reactor</u>) whose fuel is in fluid form.
Fluidized Bed Reactor	A reactor design in which the fuel ranges in size from small particles to pellets. Although the <u>fuel</u> particles are solid, their entire mass behaves like a fluid because a stream of liquid or gas <u>coolant</u> keeps them moving.
Flux (Neutron)	A term used to express the intensity of neutron radiation. The number of neutrons passing through a unit area in unit time. For neutrons of given energy, the product of neutron density with speed.
Food Chain	The pathways by which any material (such as radioactive material from fallout) passes from the first absorbing organism through plants and animals to man.
Frit, Frits	Chemically complex glasses used in a ground condition to incorporate further chemicals in a resulting final composite glass product.
FRP	<u>Fuel Reprocessing Plant</u> . (See Fuel Reprocessing.)
Fuel	Fissionable material used or usable to produce energy in a reactor. Also applied to a mixture, such as natural uranium, in which only part of the atoms are readily fissionable, if the mixture can be made to sustain a chain reaction. (See <u>fissionable material</u> .)
Fuel Bundle	A designed array of fuel elements in the holder or in a moderation matrix.
Fuel Cycle	The series of steps involved in supplying fuel for nuclear power reactors. It includes mining, refining, enrichment, fabrication of fuel elements, use in a reactor, chemical processing to recover the fissionable material remaining in the spent fuel, re-enrichment of

the recovered uranium, refabrication into new fuel elements, transportation of materials between these various stages, and management of radioactive wastes.

Fuel Element	A rod, tube, plate, or other mechanical shape or form into which nuclear fuel is fabricated for use in a reactor. (Not to be confused with element.) (See <u>nuclear reactor</u> .)
Fuel Reprocessing	The processing of reactor fuel to recover the unused fissionable material. (See <u>recycling</u> , <u>spent fuel</u> .)
Fused-salt Reactor	A type of reactor that uses molten salts of uranium for both fuel and coolant.
Fusion	The formation of a heavier nucleus from two lighter ones (such as hydrogen isotopes), with the attendant release of energy (as in a hydrogen bomb). (Compare <u>fission</u> ; see <u>nuclear reaction</u> , <u>Sherwood</u> .)
Fusion Weapon	An atomic weapon using the energy of nuclear fusion, such as a <u>hydrogen bomb</u> .
Gamma Rays	High-energy, short-wavelength electromagnetic radiation. Gamma radiation frequently accompanies alpha and beta emissions and always accompanies fission. Gamma rays are very penetrating and best stopped or shielded against by dense materials, such as lead or depleted uranium. Gamma rays are essentially similar to X-rays, but are usually more energetic, and are nuclear in origin. (See <u>decay radioactive</u> , <u>photon</u> .)
Gas Centrifuge Process	A method of isotopic separation in which heavy gaseous atoms or molecules are separated from light ones by centrifugal force. (See <u>isotope separation</u> .)
Gas-cooled Reactor	A nuclear reactor in which a gas is the coolant.
Gaseous Diffusion (plant)	A method of isotopic separation based on the fact that gas atoms or molecules with different masses will diffuse through a porous barrier (or membrane) at different rates. The method is used by the AEC to separate uranium-235 from uranium-238; it requires large gaseous-diffusion plants and enormous amounts of electric power. (See <u>cascade</u> , <u>isotope separation</u> , <u>uranium hexafluoride</u> .)

Genetic effects of Radiation	Radiation effects that can be transferred from parent to offspring. Any radiation caused changes in the genetic material of sex cells. (Compare <u>radiomutation</u> , <u>somatic effects of radiation</u> .)
Genetically Significant Dose (GSD).	The gonadal dose which, if received by every member of the population, would be expected to produce the same total genetic effect on the population as the sum of the individual doses that are actually received. It is not a forecast of predictable adverse effects on any individual person or his/her unborn children.
Geosphere	The solid mass of earth, as distinct from the atmosphere or hydrosphere.
GESMO	<u>General environmental impact statement for mixed oxide fuels (recycle plutonium in light-water-cooled reactors.)</u>
Half-life	The time in which half the atoms of a particular radioactive substance disintegrate to another nuclear form. Measured half-lives vary from millionths of a second to billions of years. (See <u>decay</u> , <u>radioactive</u> .)
Half-life Biological	(See <u>biological half-life</u> .)
Half-life, Effective	The time required for a radionuclide contained in a biological system, such as a man or an animal, to reduce its activity by half as a combined result of radioactive decay and biological elimination. (Compare <u>biological half-life</u> ; see <u>half-life</u> .)
Head End of the Fuel Cycle	Mining, milling, enrichment, and fabrication of UO ₂ fuel.
Health Physics	The science concerned with recognition, evaluation, and control of health hazards from <u>ionizing radiation</u> .
Heat Exchanger	Any device that transfers heat from one fluid (liquid or gas) to another or the environment.
Heat sink	Anything that absorbs heat; usually part of the environment, such as the air, a river, or outer space.
Heavy Hydrogen	Another name for <u>deuterium</u>

Heavy Water (D ₂ O)	Water containing significantly more than the natural proportion (one in 6500) of heavy hydrogen (deuterium) atoms to ordinary hydrogen atoms. Heavy water is used as a moderator in some reactors because of its effectiveness in slowing down neutrons and because of its low cross section for absorption of neutrons.
Heavy-water-moderated Reactor	A reactor that uses <u>heavy water</u> as its moderator. Heavy water is an excellent moderator and thus permits the use of inexpensive natural (unenriched) uranium as a fuel.
HEPA	<u>High efficiency particulate air filters</u> . Pleated fiberglass filters with high surface area and small pore size designed to remove aerosols with a minimum efficiency of 99.97% for 0.3 micrometer particles.
Heterogeneous Reactor	A reactor in which the fuel is separate from the moderator and is arranged in discrete bodies, such as <u>fuel elements</u> . Most reactors are heterogeneous. (Compare <u>homogeneous reactor</u> .)
High-level liquid Wastes	Those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated waste from subsequent extraction cycle, or equivalent, in a facility for reprocessing reactor fuels.
High-level Wastes (HLW)	Radioactive wastes including high-level liquid waste, and spent fuel, -- if it is declared to be waste -- or any equivalent radioactive waste material.
High-temperature gas-cooled reactor (HTGR)	A reactor in which the temperature is great enough to permit generation of mechanical power at good efficiency using gas as the coolant.
Homogeneous Reactor	A reactor in which the fuel is mixed with or dissolved in the moderator or coolant. Example: a fused-salt reactor. (Compare <u>heterogeneous reactor</u> .)
Hot	Highly radioactive
Hot Spot	A surface area of higher-than-average radioactivity. Also a part of a <u>fuel element</u> surface that has become overheated.

Hulls	The remnant outer casing or solid cladding waste of fuel bundles after most of the fuel pellets have been removed by nitric acid leaching.
Hydrogen Bomb	A nuclear weapon that derives its energy largely from fusion. (See <u>thermonuclear reaction</u> .)
Immobilization	Treatment and/or emplacement of the wastes so as to impede their movement.
Indirect-cycle Reactor System	A reactor system in which a heat exchanger transfers heat from the reactor <u>coolant</u> to a second fluid which then drives a turbine. (Compare <u>closed-cycle reactor system</u> , <u>direct-cycle reactor system</u> .)
Intermediate (Epithermal) Neutron	A neutron having energy greater than that of a thermal neutron but less than that of a fast neutron. The range is generally considered to be between about 0.5 and 100,000 electron volts. (Compare <u>fast neutron</u> , <u>thermal neutron</u> .)
Intermediate-level Waste	Waste other than high level waste containing concentrations and or quantities of radioactive materials requiring some kind of action to protect personnel from radiation from the materials.
Interim storage	Storage operations for which a) monitoring and human control are provided and b) subsequent action involving treatment, transportation, or final disposition is expected. Concepts for interim storage include bulk and unitized storage of solid, liquid, and gaseous wastes. Alternative interim storage technologies include: • tank storage of liquids • canister storage in air-cooled vaults • spent fuel storage in water basins.
Internal Radiation	Radiation from a source within the body (as a result of deposition of radionuclides in body tissues).
Ion	An atom or molecule that has lost or gained one or more electrons. By this ionization it becomes electrically charged. Examples: an alpha particle, which is a helium atom

minus two electrons; a proton, which is a hydrogen atom minus its electron.

Ionizing Radiation

Any radiation displacing electrons from atoms or molecules, thereby producing ions. Examples: alpha, beta, gamma radiation, Ionizing radiation may produce severe skin or tissue damage. (See radiation, radiation burn, radiation illness.)

Ion Exchange

A chemical process involving the reversible interchange of various ions between a solution and a solid material, usually a plastic or a resin. It is used to separate and purify chemicals, such as fission products, rare earths, etc., in solutions.

Irradiation

Exposure to radiation, as in a nuclear reactor (See spent fuel.)

Isobar

One of two or more nuclides having about the same atomic mass but different atomic numbers, hence different chemical properties.

Example: $^{14}_6\text{C}$, $^{14}_7\text{N}$ and $^{14}_8\text{O}$ are isobars.

(Compare isotope).

Isointensity Contours

Imaginary lines on the surface of the ground or water, or lines drawn on a map, joining points in a radiation field which have the same radiation intensity at a given time.

Isolation

A term encompassing both final storage and/or disposal in geologic formations.

Isotone

One of several nuclides having the same number of neutrons but a different number of protons in their nuclei.

Example: potassium-39 ($^{39}_{19}\text{K}$) and calcium-40 ($^{40}_{20}\text{Ca}$) are isotones. (Compare isotope).

Isotope

One of two or more atoms with the same atomic number (the same chemical element) but with different atomic weights. An equivalent statement is that the nuclei of isotopes have the same number of protons but different numbers

of neutrons. Thus, $^{12}_6\text{C}$, $^{13}_6\text{C}$, and $^{14}_6\text{C}$ are isotopes of the element carbon, the subscripts

denoting their common atomic numbers, the superscripts denoting the differing mass numbers, or approximate atomic weights. Isotopes usually have very nearly the same chemical properties, but somewhat different physical properties. (Compare isobar, isotone, nuclide; see radioisotope.)

Isotope Separation

The process of separating isotopes from one another, or changing their relative abundances, as by gaseous diffusion or electromagnetic separation. Isotope separation is a step in the isotopic enrichment process.

Isotopic Enrichment

A process by which the relative abundances of the isotopes of a given element are altered, thus producing a form of the element which has been enriched in one particular isotope. Example: enriching natural uranium in the uranium-235 isotope. (See enriched material, gaseous diffusion.)

Kilo

A prefix that multiplies a basic unit by 1000.

Kiloton Energy

The energy of a nuclear explosion which is equivalent to that of an explosion of 1000 tons of TNT. (See TNT equivalent, yield.)

Lanthanide Series

The series of elements beginning with lanthanum, Element No. 57, and continuing through lutetium, Element No. 71, which together occupy one position in the Periodic Table of the elements. These are the "rare earths," which all have chemical properties similar to lanthanum. They also are called the "lanthanides." (Compare actinide series; see rare earths.)

Lattice

An orderly array or pattern of nuclear fuel elements and moderator in a reactor or critical assembly. Also, the arrangement of atoms in a crystal.

Leach Rate

The rate of extraction of a material per unit area by contacting it to a solvent (water, acids, or alkalies).

Leakage

In nuclear engineering, the escape of neutrons from a reactor core. Leakage lowers a reactor's reactivity. (See neutron economy.)

LET	<u>Linear Energy Transfer</u> -- average energy locally imparted to a medium per unit length of charged particle travel.
Licensed Material	<u>Source material</u> , <u>special nuclear material</u> , or <u>by-product material</u> received, possessed, used or transferred under a general or special license issued by the NRC or a state.
Light Hydrogen	Ordinary hydrogen.
Light Water	Ordinary water (H_2O), as distinguished from <u>heavy water</u> (D_2O).
LLW	<u>Low-level waste</u> (containing minimal transuranic elements).
Long-lived Nuclides	Radioactive isotopes with half-lives of several years or greater. Most nuclides of interest to waste management have half-lives on the order of tens to millions of years (^{239}Pu - 24,400 years; ^{99}Tc - 2.1×10^5 years; ^{129}I - 1.6×10^7 years).
Low-level Waste	Wastes containing types and concentrations of radioactivity such that shielding to prevent personnel exposure is not required.
Low Population Zone	An area of low population density sometimes required around a nuclear installation. The number and density of residents is of concern in providing, with reasonable probability, that effective <u>protection</u> measures can be taken if a serious accident should occur. (See <u>exclusion</u> area.)
LMFBR	<u>Liquid-Metal Fast Breeder Reactor</u> .
LWR	<u>Light-water Reactor</u> .
LWR Fuel Cycle	Light-water reactor fuel cycle. A system of operation and facilities for the preparation, utilization, and reconstitution of the LWR fuel. The main operations involve: 1. Mining of ore; 2. Milling (ore concentration and purification); 3. Conversion of U_3O_8 to UF_6 ; 4. Enrichment; 5. Conversion of UF_6 to UO_2 ; 6. Fuel fabrication; 7. Irradiation in a reactor, and 8. Reprocessing.
Mass Number	The sum of the neutrons and protons in a <u>NUCLEUS</u> . It is the nearest whole number to an atom's <u>atomic weight</u> . For instance, the mass number of uranium-235 is 235.

Matter	The substance of which a physical object is composed. All materials in the universe have the same inner nature, that is, they are composed of atoms, arranged in different (and often complex) ways; the specific atoms and the specific arrangements identify the various materials. (See <u>atom</u> and <u>element</u> .)
Maximum Credible Accident	The most serious reactor accident that can reasonably be imagined from any adverse combination of equipment malfunction, operating errors, and other foreseeable causes. The term is used to analyze the safety characteristics of a reactor. Reactors are designed to be safe even if a maximum credible accident should occur.
Maximum Permissible Concentration (MPC)	The amount of radioactive material in air, water, or food which might be expected to result in a <u>maximum permissible dose</u> to persons consuming them at a standard rate of intake. (See <u>radiation protection guide, radioactivity</u> .)
Maximum Permissible Dose	That dose of ionizing radiation established by competent authorities as an amount below which there is no reasonable expectation of risk to human health, and which at the same time is somewhat below the lowest level at which a definite hazard is believed to exist. An obsolescent term. (See <u>radiation protection guide</u> .)
Mega	A prefix that multiplies a basic unit by one million.
Megaton Energy	The energy of a nuclear explosion which is equivalent to that of an explosion of one million tons (or 1000 kilotons) of TNT. (See <u>TNT equivalent, yield</u> .)
Megawatt-day Per Ton	A unit used for expressing the burnup of fuel in a reactor; specifically, the number of megawatt-days of heat output per metric ton of fuel in the reactor. (See <u>burnup</u> .)
MeV	One million electron volts.
MFRP	<u>Midwest Fuel Recovery Plant</u> : a General Electric Reprocessing Plant located at Morris, Ill. that was built but not operated.
Micro	A prefix that divides a basic unit by one million.

Micromicro	(See Pico)
Millfeed	The ore and other material introduced into the milling process.
Milli	A prefix that divides a basic unit by one thousand.
Moderator	A material, such as ordinary water, heavy water or graphite, used in a <u>reactor</u> to slow down high-velocity neutrons, thus increasing the likelihood of further <u>fission</u> . (Compare <u>reflector</u> ; see <u>absorber</u> , <u>thermal neutrons</u> .)
Molecule	A group of atoms held together by chemical forces. The atoms in the molecule may be identical, as in H ₂ , S ₂ , and S ₈ , or different, as in H ₂ O, and CO ₂ . A molecule is the smallest unit of matter which can exist by itself and retain all of its chemical properties. (Compare atom, ion.)
Molten Salt Reactor	A <u>fused-salt reactor</u> .
MOX	<u>Mixed oxide reactor fuel</u> (containing both plutonium and uranium.)
MPCA	<u>Maximum Permissible Concentration in air</u> . See <u>Maximum Permissible Concentration</u>
MPCW	<u>Maximum Permissible Concentration in water</u> . See <u>Maximum Permissible Concentration</u> .
MT	<u>Metric tons</u> .
MTU	<u>Metric Ton of Uranium fuel</u> .
Multiplication Factor	(Symbol K) The ratio of the number of neutrons present in a reactor in any one neutron generation to that in the immediately preceding generation. Criticality is achieved when this ratio is equal to one. The "infinite" multiplication factor is the ratio in a theoretical system from which there is no leakage, that is, a reactor of infinite size; for an actual reactor (from which leakage does occur), the term effective multiplication factor, which is the ratio based on neutrons available after leakage, is commonly used. (See generation time, leakage, neutron, neutron, reactivity.)
Mutation	A permanent change in the characteristics of an offspring from those of its parents. (Compare <u>radiomutation</u> .)

Nano	A prefix that divides a basic unit by one billion (10 ⁹)
Natural Circulation Reactor	A reactor in which the coolant (usually water) is made to circulate without pumping, that is, by natural convection resulting from the different densities of its cold and reactor-heated portions.
Natural Radiation	<u>background radiation</u>
Natural Uranium	Uranium as found in nature, containing 0.7% of ²³⁵ U, 99.3% of ²³⁸ U, and a trace of ²³⁴ U. It is also called <u>normal uranium</u> . (See <u>uranium</u> .)
Neptunium Series (sequence)	The series of nuclides resulting from the radioactive decay of the man-made nuclide, neptunium-237. Many other man-made nuclides decay into this sequence. The end-product of the series is stable bismuth-209, which is the only nuclide in the series that occurs in nature. (See <u>decay</u> , <u>radioactive</u> .)
Neutralized Waste	Liquid nuclear waste which was originally a nitric acid solution has been treated subsequently by Sodium Hydrozide so as to suppress its corrosiveness to the container wall.
Neutron	An uncharged <u>elementary particle</u> , with a mass slightly greater than that of the proton, and found in the nucleus of every atom heavier than hydrogen. A free neutron is unstable and decays with a half-life of about 13 minutes into an electron, proton, and neutrino. Neutrons sustain the fission chain reaction in a <u>nuclear reactor</u> . (See <u>fast neutron</u> , <u>intermediate neutron</u> , and <u>thermal neutron</u> .)
Neutron Economy	The degree to which neutrons in a <u>reactor</u> are used for desired ends instead of being lost by leakage or nonproductive absorption. The ends may include propagation of the chain reaction, converting fertile to fissionable material, producing isotopes, or research. (See <u>leakage</u> , <u>reactivity</u> .)
Non-high Level Waste	Intermediate- or low-level waste.
Normal Uranium	<u>natural uranium</u>
NFS	<u>Nuclear Fuel Services</u> . A reprocessing plant located at West Valley, N.Y. that operated from 1966 to 1972.

NRC	Nuclear Regulatory Commission; a federal agency, successor to regulatory functions of the Atomic Energy Commission.
Nuclear Energy	The energy liberated by a nuclear reaction (fission or fusion) or by radioactive decay. (See <u>decay</u> , <u>radioactive</u> ; <u>fission</u> ; <u>fusion</u> ; <u>nuclear explosive</u> ; <u>nuclear reactor</u> .)
Nuclear Fission	(See <u>fission</u> .)
Nuclear Fusion	(See <u>fusion</u> .)
Nuclear Power Plant	Any device, machine, or assembly that converts nuclear energy into some form of useful power, such as mechanical or electrical power. In a nuclear electric power plant, heat produced by a <u>reactor</u> is generally used to make steam to drive a turbine that in turn drives an electric generator.
Nuclear Reaction	A reaction involving a change in an atomic nucleus, such as <u>fission</u> , <u>fusion</u> , <u>neutron capture</u> , or <u>radioactive decay</u> , as distinct from a chemical reaction, which is limited to changes in the electron structure surrounding the nucleus.
Nuclear Reactor	A device in which a fission <u>chain reaction</u> can be initiated, maintained, and controlled. Its essential component is a <u>core</u> with fissionable <u>fuel</u> . It usually has a <u>moderator</u> , a <u>reflector</u> , <u>shielding</u> , <u>coolant</u> , and control mechanisms. Sometimes called an atomic "furnace," it is the basic machine of <u>nuclear energy</u> . (See <u>fission</u> .)
Nuclear Superheating	Superheating the steam produced in a reactor by using additional heat from a reactor. Two methods are commonly employed;: recirculating the steam through the same core in which it is first produced (integral superheating) or passing the steam through a second and separate reactor. (See <u>superheating</u> .)
Nuclei	Plural of <u>nucleus</u> .
Nucleon	A constituent of an atomic nucleus, that is, a <u>proton</u> or a <u>neutron</u> .
Nucleus	The positively charged center of an atom.
Nuclide	A general term applicable to all atomic forms of the elements. The term is often erroneously

	used as a synonym for "isotope," which properly has a more limited definition. Whereas isotopes are the various forms of a single element (hence are a family of nuclides) and all have the same atomic number an(number of protons), nuclides comprise all the isotopic forms of all the elements. Nuclides are distinguished by their atomic number, atomic mass, and energy state.
Off Gases	Volatile substances released during operation of fuel cycle from system components such as milling, reactor, and reprocessing plants. Off gases may be subsequently scrubbed and filtered before release.
Open-Cycle Reactor System	A reactor system in which the coolant passes the reactor core only once and is then discarded. (Compare closed-cycle reactor system.)
Operations	Broad classification of waste management activities in terms of their basic function (e.g., waste storage, treatment, transportation or disposal.)
Orange Oxide	Uranium trioxide.
Orbit	The region occupied by an electron as it moves about the nucleus of an atom.
Organic-cooled Reactor	A reactor that uses organic chemicals, such as mixtures of polyphenyls (diphenyls and terphenyls) as coolant.
Overpack	Secondary (or additional) external containment for packaged nuclear waste.
Overpressure	The transient pressure over and above atmospheric pressure caused by a shock wave from a nuclear explosion.
Parasitic Capture	Any absorption (as in a reactor) of neutrons in reactions which do not cause further fission or the production of new fissionable material. In a reactor the process is undesirable. (See <u>absorption</u> , <u>capture</u> , <u>neutron economy</u> .)
Parent	A radionuclide that upon radioactive decay or disintegration yields a specific nuclide (the daughter), either directly or as a later member of a radioactive series. (See <u>daughter</u> , <u>radioactive series</u> .)
Partition	The separation of uranium and plutonium from the spent fuel solution through the solvent

extraction process. This pertains to reprocessing (Cf. LWR Fuel Cycle).

Partitioning

The process of separating liquid waste into two or more fractions. In this report, partitioning is used specifically with reference to the removal of certain radioisotopes from the waste in order to facilitate subsequent waste storage and disposal. "Isotope mining" is used to describe the fractionation of waste when radioisotopes are extracted and used in other applications.

Pebble Bed Reactor

A reactor in which the fissionable fuel (and sometimes also the moderator) is in the form of packed or randomly placed pellets, which are cooled by gas or liquid.

Period

The time required for one cycle of a regularly repeated series of events. In a nuclear reactor, it is the time required for the power level to change by the factor 2.718, which is known as e (the base of natural logarithms). (See Periodic Table.)

Photon

The carrier of a quantum of electromagnetic energy. Photons have an effective momentum but no mass or electrical charge. (See radiation, quantum.)

Pico

A prefix that divides a basic unit by one trillion (10^{12}). Same as micromicro.

Pile

Old term for nuclear reactor. This name was used because the first reactor was built by piling up graphite blocks and natural uranium.

Plutonium

[Symbol Pu] A heavy, radioactive, metallic element with atomic number 94. Its most important isotope is fissionable plutonium-239, produced by neutron irradiation of uranium-238. It is used for reactor fuel and in weapons.

Poison

Any material of high absorption cross section that absorbs neutrons unproductively and hence removes them from the fission chain

reaction in a reactor, decreasing its reactivity. (Compare burnable poison.)

Pool Reactor

A reactor in which the fuel elements are suspended in a pool of water that serves as the reflector, moderator, and coolant. Popularly called a swimming pool reactor, it is usually used for research and training. (Compare tank reactor.)

Power Density	The rate of heat generated per unit volume of a <u>reactor</u> core. (See <u>specific power</u> .)
Power Reactor	A reactor designed to produce useful nuclear power is distinguished from reactors used primarily for research or for producing radiation or fissionable materials. (Compare production reactor, research reactor.)
Pressure-tube Reactor	A reactor in which the fuel elements are located inside tubes containing coolant circulating at high pressure. The tube assembly is surrounded by a tank containing the moderator at low pressure.
Pressure Vessel	A strong-walled container housing the core of most types of power reactors; it usually also contains moderator, reflector, thermal shield, and control rods. (Compare <u>containment vessel</u> .)
Pressurized Water Reactor (PWR)	A power reactor in which heat is transferred from the core to a heat exchanger by water kept under high pressure to achieve high temperature without boiling in the primary system. Steam is generated in a secondary circuit. Many reactors producing electric power are pressurized water reactors.
Primary Wastes	As-generated forms and quantities of wastes.
Process Heat Reactor	A reactor that produces heat for use in manufacturing processes.
Production Reactor	A reactor designed primarily for large-scale production of plutonium-239 by neutron <u>irradiation</u> of uranium-238. Also a reactor used primarily for the production of radioactive isotopes. (Compare power reactor, <u>research reactor</u> .)
Prompt Criticality	The state of a reactor when the fission chain reaction is sustained solely by prompt neutrons, that is, without the help of delayed neutrons. (See criticality.)
Prompt Neutron	Neutrons that are emitted immediately following nuclear fissions, as distinct from delayed neutrons, which are emitted for some time after fission has occurred. Prompt neutrons comprise more than 99% of fission neutrons.
Provisional Storage	Storage in a planned disposal medium while disposal feasibility is assessed.

Pulsed Reactor	A type of research reactor with which repeated short, intense surges of power and radiation can be produced. The <u>neutron flux</u> during each surge is much higher than could be tolerated during a steady-state operation.
Purex	Abbreviation for plutonium-uranium extraction.
Purex Process	An aqueous reprocessing method which consists of the following steps: (1) Solvent extraction which effects the fission products and partition of uranium and plutonium, (2) purification of uranium and plutonium, (3) conversion of uranium to UF_6 and plutonium to PuO_2 .
Purex Redox	Redox is the abbreviation for reduction and oxidation. This implies that it is a Purex process which involves both the reduction and oxidation steps in the solvent extraction.
PWR	<u>P</u> ressurized <u>W</u> ater <u>R</u> eactor.
Quad	A unit equal to 10^{15} Btu.
Quantum	Unit quantity of energy according to the quantum theory. It is equal to the product of the frequency of radiation of the energy in sec^{-1} and 6.6256×10^{27} erg-sec. The photon carries a quantum of electromagnetic energy. (See electromagnetic radiation, radiation.)
Quality Factor	The factor by which absorbed dose is to be multiplied to obtain a quantity that expresses on a common scale, for all ionizing radiations, the irradiation incurred by exposed persons. (See dose equivalent, distribution factor.)
Rad	(Acronym for <u>r</u> adiation <u>a</u> bsorbed <u>d</u> ose). The basic unit of absorbed dose of ionizing radiation. A dose of one rad means the absorption of 100 ergs of radiation energy per gram of absorbing material. (Compare rem, roentgen; see absorbed dose.)
Radiation	The emission and propagation of <u>energy</u> through matter or space by means of electromagnetic disturbances which display both wave-like and particle-like behavior; in this context the "particles" are known as <u>photons</u> . Also, the energy so propagated. The term has been extended to include streams of fast-moving <u>particles</u> (alpha and beta particles, free neutrons, cosmic radiation, etc.). Nuclear

	radiation is that emitted from atomic nuclei in various <u>nuclear reactions</u> , including alpha, beta and gamma radiation and neutrons. (See <u>electromagnetic radiation</u> , <u>ionizing radiation</u> , <u>quantum</u> .)
Radiation Accidents	Accidents resulting in the spread of radioactive material or in the exposure of individuals to radiation.
Radiation Area	Any accessible area in which the level of radiation is such that a major portion of an individual's body could receive in any one hour a dose in excess of 5 millirem, or in any 5 consecutive days a dose in excess of 150 millirem. (See <u>absorbed dose</u> , <u>rem</u> .)
Radiation Biology	(See <u>radiobiology</u> .)
Radiation Burn	Radiation damage to the skin. Beta burns result from skin contact with or exposure to emitters of beta particles. Flash burns result from sudden thermal radiation. (See beta particles, flash burn, ionizing radiation, thermal burn.)
Radiation Chemistry	The branch of chemistry that is concerned with the chemical effects, including decomposition, of energetic radiation or particles on matter. (Compare <u>radiochemistry</u> .)
Radiation Damage	A general term for the harmful effects of radiation on matter.
Radiation Illness	An acute organic disorder that follows exposure to relatively severe doses of ionizing radiation. It is characterized by nausea, vomiting, diarrhea, blood cell changes, and in later stages by hemorrhage and loss of hair. (See ionizing radiation.)
Radiation Monitoring	Continuous or periodic determination of the amount of radiation present in a given area. (See <u>monitor</u> .)
Radiation Protection	Legislation and regulations to protect the public and laboratory or industrial workers against radiation. Also measures to reduce exposure to radiation. (See <u>radiation standards</u> .)
Radium	A radioactive metallic element with atomic number 88. As found in nature, the most

common isotope has an atomic weight of 226. It occurs in minute quantities associated with uranium in pitchblende, carnotite and other minerals; the uranium decays to radium in a series of alpha and beta emissions. By virtue of being an alpha- and gamma-emitter, radium is used as a source of luminescence and as a radiation source in medicine and radiography.

Radon

A radioactive element, one of the heaviest gases known. Its atomic number is 86, and its atomic weight is 222. It is a daughter of radium in the uranium radioactive series.

Rare Earths

A group of 15 chemically similar metallic elements, including Elements 57 through 71 on the Periodic Table of the Elements, also known as the Lanthanide Series. (See lanthanide series.)

Reactivity

A measure of the departure of a nuclear reactor from criticality. It is about equal to the effective multiplication factor minus one and is thus precisely zero at criticality. If there is excess reactivity (positive reactivity), the reactor is supercritical and its power will rise. Negative reactivity (subcriticality) will result in a decreasing power level. (See criticality, dollar, excess reactivity, multiplication factor, subcritical assembly, supercritical reactor.)

Reactor

(See nuclear reactor.)

Recycling

The reuse of fissionable material, after it has been recovered by chemical processing from spent or depleted reactor fuel, reenriched, and then refabricated into new fuel elements. (See fuel cycle, fuel reprocessing, spent fuel.)

Reflector

A layer of material immediately surrounding a reactor core which scatters back or reflects into the core many neutrons that would otherwise escape. The returned neutrons can then cause more fissions and improve the neutron economy of the reactor. Common reflector materials are graphite, beryllium, and natural uranium. (Compare moderator.)

Relative Biological Effectiveness (RBE)

A factor used to compare the biological effectiveness of different types of ionizing radiation. It is the inverse ratio of the amount of absorbed radiation, required to

	produce a given effect, to a standard (or reference) radiation required to produce the same effect. (See <u>absorbed dose</u> , <u>distribution factor</u> , <u>quality factor</u> , <u>rad</u> , <u>rem</u> .)
Radioactivity	Activity is a measure of the rate at which a material is emitting nuclear radiations, and is usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time. The standard unit of activity is the curie (Ci), which is equal to 3.7×10^{10} disintegrations per second. The words "activity" and "radioactivity" are often used interchangeably.
Radioactivity Concentration Guide (RCG)	The concentration of radioactive material in an environment which would result in doses equal, over a period of time, to those in the <u>Radiation Protection Guide</u> . This Federal Radiation Council term replaces the former <u>maximum permissible concentration</u> .
Radiobiology	The body of knowledge and the study of the principles, mechanisms, and effects of ionizing radiation on living matter.
Radiochemistry	The body of knowledge and the study of the chemical properties and reactions of radioactive materials.
Radioecology	The body of knowledge and the study of the effects of radiation on species of plants and animals in natural communities.
Radiogenic	Of radioactive origin; produced by radioactive <u>transformation</u> . (See <u>decay</u> , <u>radioactive</u> ; <u>transmutation</u> .)
Radioisotope	A radioactive isotope. An unstable isotope of an element that decays or disintegrates spontaneously, emitting radiation. More than 1300 natural and artificial radioisotopes have been identified. (See <u>decay</u> , <u>radioactive</u> ; <u>isotope</u> .)
Radiolysis	The dissociation (or decomposition) of molecules by radiation. Example: A small proportion of water in a reactor core dissociates into hydrogen and oxygen during operation of the reactor.
Radiomutation	A permanent transmissible change in form, quality, or other characteristic of a cell or offspring from the characteristics of its parent, due to radiation exposure. (See <u>genetic effects of radiation</u> , <u>mutation</u> .)

Radionuclide	A <u>radioactive nuclide</u> .
Radiation Shielding	Reduction of radiation by interposing a shield of absorbing material between any radioactive source and a person, laboratory area, or radiation-sensitive device. (See <u>absorber</u> , <u>shield</u> .)
Radiation Standards	Exposure standards, permissible concentrations, rules for safe handling, regulations for transportation, regulations for industrial control of radiation, and control of radiation exposure by legislative means. (See <u>radiation protection</u> .)
Radio	A prefix denoting <u>radioactivity</u> or a relationship to it, or a relationship to radiation.
Radioactivation	Activation.
Radioactive	Exhibiting radioactivity or pertaining to radioactivity.
Radioactive Chain	A <u>radioactive series</u> .
Radioactive Contamination	Deposition of radioactive material in any place where it may harm persons, spoil experiments, or make products or equipment unsuitable or unsafe for some specific use. The presence of unwanted radioactive matter. Also radioactive material found on the walls of vessels in used-fuel processing plants, or radioactive material that has leaked into a reactor coolant. Often referred to only as <u>contamination</u> . (Compare <u>background radiation</u> ; see <u>decontamination</u> .)
Radioactive Decay	Disintegration of the nucleus of an unstable nuclide by spontaneous emission of charged particles and/or photons.
Radioactive Half-Life	(See <u>half-life</u> .)
Radioactive Isotope	A <u>radioisotope</u> .
Radioactive Series	A succession of nuclides, each of which transforms by radioactive disintegration into the next until a stable nuclide results. The first member is called the <u>parent</u> , the intermediate members are called <u>daughters</u> , and the final stable member is called the <u>end product</u> . (See <u>decay</u> , <u>radioactive</u> .)

Radioactive Waste	(See waste, radioactive.)
Rem	(Acronym for <u>roentgen equivalent man</u> .) The unit of dose of any ionizing radiation which produces the same biological effect as a unit of absorbed dose of ordinary X rays. The RBE dose (in rems) = RBE x absorbed dose (in rads). (Compare curie, roentgen.)
Repository	A location containing wastes in storage or disposal.
Reprocessing	<u>Fuel reprocessing</u> .
Research Reactor	A reactor primarily designed to supply neutrons or other ionizing radiation for experimental purposes. It may also be used for training, materials testing, and production of radioisotopes. (Compare <u>experimental reactor</u> , <u>power reactor</u> , <u>production reactor</u> , <u>test reactor</u> .)
Retrievability	Capability to remove waste from its place in final storage. The method and rate of removal and the subsequent location of the waste must satisfy retrievability criteria.
Rod	A relatively long, slender body of material used in or in conjunction with a nuclear reactor. It may contain fuel, absorber, or material in which activation or transmutation is desired. (See <u>control rod</u> .)
Roentgen	(Abbreviation r). A unit of exposure to ionizing radiation. It is that amount of gamma or x rays required to produce ions carrying 1 electrostatic unit of electrical charge (either positive or negative) in 1 cubic centimeter of dry air under standard conditions. Named after Wilhelm Roentgen, German scientist who discovered X rays in 1895.
RSSF	<u>Retrievable Surface Storage Facility</u> . A facility or system for interim storage of high level waste. Allows placement of packaged waste in controlled area mausoleums for eventual retrieval for reprocessing or for retrieval for permanent disposal.
Safe Guards	Precautionary regulations or measures to ward off possible nuclear theft, sabotage and terrorist activities.

Salt Cake	The solid residue resulting from a concentration of high-level waste in underground waste storage tanks.
Scram	The sudden shutdown of a nuclear reactor, usually by rapid insertion of the <u>safety rods</u> . Emergencies or deviations from normal reactor operation cause the reactor operator or automatic control equipment to scram the reactor.
Secondary Wastes	Forms and quantities of all wastes that result from applying waste treatment technologies to primary wastes.
Seed (and Blanket) Core	A reactor core which includes a relatively small volume of highly <u>enriched uranium</u> (the seed) surrounded by a much larger volume of <u>natural uranium</u> or thorium (the blanket). As a result of fissions in the seed, neutrons are supplied to the blanket where more fission takes place. In this way, the blanket is made to furnish a substantial fraction of the total power of the reactor. Also called a <u>spiked core</u> .
Sherwood	The Atomic Energy Commission program for research in controlled thermonuclear reactions.
Short-lived Nuclides	Radioactive isotopes with half-lives no greater than about 30 years, e.g., ^{137}Cs and ^{90}Sr .
Single-cycle Reactor System	A <u>direct-cycle reactor system</u> .
Slow Neutron	A <u>thermal neutron</u> .
Sludge	A wet solid settling out from a pump sump, e.g., the solid waste which comes from liquid effluents from process clean up operations, generated during enrichment process. The sludge thus collected contains soil runoff from ground water, small quantities of precipitated metals, various suspended solids, and a detectable amount of radioactivity.
SNM	<u>Special Nuclear Material</u> . -- In atomic energy law, this term refers to plutonium-239, uranium-233, uranium containing more than the natural abundance of uranium-235, or any material artificially enriched in any of these substances.
Sodium-graphite Reactor	A reactor that uses liquid sodium as coolant and graphite as moderator.
Solidification	Conversion of radioactive waste to a dry, stable solid.

Solvent Extract	The extracted metal salt (uranium) from the ore concentrates, through the selective transfer of the desired metal (Uranium) salt from the aqueous liquor into an immiscible (two liquids that do not mix and form more than one phase when brought together) organic solvent (tributyl-phosphate in hexane) after initial stirring followed by phase separation. This is also used in the <u>partition</u> process.
Somatic Effects of Radiation	Effects of radiation limited to the exposed individual, as distinguished from genetic effects (which also affect subsequent, unexposed generations). Large radiation doses can be fatal. Smaller doses may make the individual noticeably ill, may merely produce temporary changes in blood-cell levels detectable only in the laboratory, or may produce no detectable effects whatever. Also called physiological effects of radiation. (Compare <u>genetic effects of radiation</u> ; see <u>radiation illness</u> .)
Source Material	In atomic energy law any material, except <u>special nuclear material</u> , which contains 0.05% or more of uranium, thorium, or any combination of the two. (See <u>licensed material</u> , <u>special nuclear material</u> .)
SNAP	(Acronym for <u>Systems for Nuclear Auxiliary Power</u> .) An Atomic Energy Commission program to develop small auxiliary nuclear power sources for specialized space, land, and sea uses. Two approaches are deployed: the first uses heat from radioisotope decay to produce electricity directly by thermoelectric or thermionic methods; the second uses heat from small reactors to produce electricity by thermoelectric or thermionic methods or by turning a small turbine and electric generator.
Soxhlet Test	A test which pertains to the continuous extraction of a solid substance with a solvent, consisting of a distillation flask, a reflex chamber, and a cylindrical vessel fitted between them, to which a siphon system is attached.
Special Nuclear Material (SNM)	See SNM
Specific Power	The power generated in a nuclear reactor per unit mass of fuel. It is expressed in kilowatts of heat per kilogram of fuel. (See <u>power density</u> .)

Spent (Depleted) Fuel	Nuclear reactor fuel that has been irradiated (used) to the extent that it can no longer effectively sustain a chain reaction. (Compare <u>depleted uranium</u> ; see <u>burnup</u> .)
Spent Fuel Rods	Reactor fuel elements which must be replaced due to a. Swelling or bursting, b. Burn-up or depletion, c. Poisoning by fission fragments. The fissile material is not necessarily exhausted and the so-called spent fuel rods could be reprocessed subsequently.
Spill	The accidental release of radioactive material.
Spontaneous Fission	Fission that occurs without an external stimulus. Several heavy isotopes decay mainly in this manner; examples: californium-252 and californium-254. The process occurs occasionally in all fissionable materials, including uranium-235.
Spray - Calciner	A technique to convert high-level liquid radioactive waste (HLLW) generated in the reprocessing of spent power reactor fuel to a dry solid. The HLLW is pumped to an internal mixing pneumatic atomizing nozzle in the top of the heated (wall temperature 700°C) Spray, Calciner barrel. The atomized droplets are flash dried and calcined as they fall through the hot barrel.
SSCC	<u>Sealed Storage Cache Concept</u> . A concept developed as an option for RSSF and SURFF. Suggested in WASH 1539.
Stable	Incapable of spontaneous change. Not radioactive.
Stable Isotope	An isotope that does not undergo radioactive decay. (Compare radioisotope.)
Storage	Temporary placement of wastes, usually in a pool or other structure, that will allow cooling and radioactive decay in safety pending shipment to a reprocessing plant or repository.
Stress Corrosion	Chemical corrosion, such as reactor pressure vessels, that is accelerated by stress concentrations, either built into or resulting from a load.
Sub-critical Assembly	A reactor consisting of a mass of fissionable material and moderator whose effective multiplication factor is less than one and that hence cannot sustain a chain reaction. (See <u>criticality</u> , <u>multiplication factor</u> , <u>reactivity</u> .)

Subcritical Mass	An amount of fissionable material insufficient in quantity or of proper geometry to sustain a fission chain reaction. (See <u>critical mass</u> , <u>criticality</u> .)
Supercritical Mass	A mass of fuel whose effective multiplication factor is greater than one. (See <u>critical mass</u> , <u>multiplication factor</u> .)
Supercritical Reactor	A reactor in which the effective multiplication factor is greater than one; consequently a reactor that is increasing its power level. If uncontrolled, a supercritical reactor would undergo an excursion. (See <u>criticality</u> , <u>excursion</u> , <u>multiplication factor</u> .)
Superheating	The heating of a vapor, particularly saturated (wet) steam, to a temperature much higher than the boiling point at the existing pressure. This is done in power plants to improve efficiency and to reduce condensation in the turbines.
Surface Contamination	The deposition and attachment of radioactive materials to a surface. (See <u>radioactive contamination</u> .)
SURFF	<u>Spent Unreprocessed Fuel Facility</u> . A facility or system for storage of spent fuel bundles from nuclear reactors. Differs from RSSF in that SURFF emphasizes anti-proliferation features using waste-management technology.
TAD	<u>Technical Alternative Document</u> ; ERDA-76-43. Alternatives for Managing Wastes From Reactors and Post-fission Operations in the LWR Fuel Cycle.
Tails	(See <u>depleted uranium</u> .)
Tank Reactor	A reactor in which the core is suspended in a closed tank, as distinct from an open pool reactor. These are commonly used as research and test reactors. (Compare <u>pool reactor</u> .)
Target	Material subjected to particle bombardment (as in an <u>accelerator</u>) or irradiation (as in a research reactor) in order to induce a <u>nuclear reaction</u> ; also a nuclide that has been bombarded or irradiated.
TBP	<u>Tributyl Phosphate</u>
Technologies	The specified methods for implementing concepts. An example is calcination of liquid high-level waste by using a spray calciner.

Terrestrial Radiation	Radiation emitted by naturally occurring radionuclides such as potassium-40; the natural decay chains uranium-238, uranium-235, or thorium-232; or from cosmic-ray induced radionuclides in the soil.
Test Reactor	A reactor specially designed to test the behavior of materials and components under the neutron and gamma fluxes and temperature conditions of an operating reactor. (Compare experimental reactor, research reactor.)
Thermal Breeder Reactor	A breeder reactor in which the fission chain reaction is sustained by <u>thermal neutrons</u> .
Thermal (slow) Neutron	A neutron in thermal equilibrium with its surrounding medium. Thermal neutrons are those that have been slowed down by a <u>moderator</u> to an average speed of about 2200 meters per second (at room temperature) from the much higher initial speeds they had when expelled by fission. This velocity is similar to that of gas molecules at ordinary temperatures. (Compare <u>fast neutron</u> , <u>intermediate neutron</u> ; see <u>fission</u> .)
Thermonuclear Reactor	A reactor in which the fission chain reaction is sustained primarily by thermal neutrons. Most reactors are thermal reactors. (Compare fast reactor, intermediate reactor; see thermal neutron.)
Thermal Reactor	A reactor in which the fission chain reaction is sustained primarily by thermal neutrons. Most reactors are thermal reactors. (Compare <u>fast reactor</u> , <u>intermediate reactor</u> ; see <u>thermal neutron</u> .)
Thermal Shield	A layer or layers of high density material located within a reactor pressure vessel or between the vessel and the biological shield to reduce radiation heating in the vessel and the biological shield. (See <u>biological shield</u> , <u>shield</u> .)
Thorex Process	A process for the recovery of Thorium and ^{233}U from fission products by the use of tributylphosphate extraction.
Thorium	[Symbol Th] A naturally radioactive element with atomic number 90 and, as found in nature, an atomic weight of approximately 232. The fertile thorium-232 isotope is abundant and can be transmuted to fissionable uranium-233 by neutron irradiation. (See <u>fertile material</u> , <u>transmutation</u> .)

Thorium Series (sequence)	The series of nuclides resulting from the radioactive decay of thorium-232. Many man-made nuclides decay into this sequence. The end product of this sequence in nature is lead-208. (See <u>decay</u> , <u>radioactive</u> ; <u>radioactive series</u> .)
Threshold Dose	The minimum dose of radiation that will produce a detectable biological effect. (See <u>absorbed dose</u> , <u>biological dose</u> .)
TNT Equivalent	A measure of the energy released in the detonation of a <u>nuclear explosive</u> expressed in terms of the weight of TNT (the chemical explosive, trinitrotoluene) which would release the same amount of energy when exploded. It is usually expressed in kilotons or megatons. The TNT equivalence relationship is based on the fact that 1 ton of TNT releases one billion (10 ⁹) calories of energy.
Tonne	Metric ton; 1000 kilograms.
Tracer, Isotopic	An isotope of an element, a small amount of which may be incorporated into a sample of material (the carrier) in order to follow (trace) the course of the element through a chemical, biological, or physical process, and thus also follow the larger carrier. The tracer may be radioactive, in which case observations are made by measuring the radioactivity. If the tracer is stable, mass spectrometers, density measurement, or neutron activation analysis may be employed to determine isotopic composition. Tracers are also called labels or tags, and materials are said to be labeled or tagged when radioactive tracers are incorporated in them.
Transmutation	Conversion of a radioactive nucleus to another isotope by bombarding it with radiation or nuclear particles.
Transplutonium Element	An element above <u>plutonium</u> in the <u>PERIODIC TABLE</u> , that is, one with an atomic number greater than 94. (See <u>transuranic element</u> .)
Transportation	Movement of materials between sites. Intra-site movement is not considered. Includes alternative methods for packaging, handling, and transport of waste materials and plutonium compounds. Concepts include all conventional methods of land and water transport required by the waste management system.

Transuranic Element (isotope)	An element above <u>uranium</u> in the <u>PERIODIC TABLE</u> , that is, with an atomic number greater than 92. All 11 transuranic elements are produced artificially and are radioactive. They are neptunium, plutonium, americium, curium, berkelium, californium, einsteinium, fermium, mendelevium, nobelium, and lawrencium.
Transuranic Elements	Elements with mass number greater than 92. They include neptunium, plutonium, americium, and curium.
Transuranic Waste	Any waste material measured or assumed to contain more than a specified concentration (e.g., presently proposed as 10 nanocuries of alpha emitters per gram of waste) of transuranic elements.
Transuranium Element	A <u>transuranic element</u> .
Treatment	Operations intended to benefit safety or economy by changing the waste characteristics. Four basic treatment concepts are defined: • Volume reduction • Immobilization of radioactivity • Change of composition • Removal of radioactivity from the waste
Tritium	A radioactive <u>isotope</u> of hydrogen with two neutrons and one proton in the nucleus. It is man-made and is heavier than deuterium (heavy hydrogen). Tritium is used in industrial thickness gauges, and as a label in experiments in chemistry and biology. Its nucleus is a triton. (Compare <u>deuterium</u> ; see <u>hydrogen</u> .)
TRU	<u>Transuranic</u> .
Uranium	A radioactive element with the atomic number 92 and, as found in natural ores, an average atomic weight of approximately 238. The two principal natural isotopes are uranium-235 (0.7% of natural uranium), which is fissionable, and uranium-238 (99.3% of natural uranium) which is fertile. Natural uranium also includes a minute amount of uranium-234. Uranium is the basic raw material of nuclear energy. (See fertile material, fissionable material, natural uranium.)
U-235	Uranium-235.
Uranium Enrichment	(See isotopic enrichment.)

Uranium Hexafluoride	A volatile compound of uranium and fluorine. UF ₆ gas is the process fluid in the gaseous diffusion process. (See isotope separation.)
Uranium Series (sequence)	The series of nuclides resulting from the radioactive decay of uranium-238, also known as the uranium-radium series. The end product of the series is lead-206. Many man-made nuclides decay into this sequence. (See <u>decay</u> , <u>radioactive</u> .)
Uranium Tetrafluoride	A solid green compound called green salt. An intermediate product in the production of uranium hexafluoride. (See uranium hexafluoride.)
Uranium Trioxide	An intermediate product in the refining of uranium, also called orange oxide.
Vitrification	The formation of glossy or non-crystalline material out of nuclear wastes, when subjected to temperatures between 950°C and 1150°C. This technique is used for the near-term immobilization of reactor wastes.
Waste, radioactive	Equipment and materials (from nuclear operations) which are radioactive and for which there is no further use. Wastes are generally classified as high-level (having radioactivity concentration of hundreds to thousands of curies per gallon or cubic foot), low-level (in the range of 1 microcurie per gallon or cubic foot), or intermediate (between these extremes). (Compare <u>fission</u> products.)
Water boiler	A research reactor whose core consists of a small metal tank filled with uranium fuel in an aqueous solution. Heat is removed by a cooling coil in the core. Not to be confused with <u>boiling water reactor</u> .
Wet Criticality	Reactor criticality achieved with the coolant present. (Compare <u>dry criticality</u> .)
WSEP	Abbreviation for <u>waste solidification evaluation program</u> , which includes calcination.
Wt%	Weight percent.
Yield	The total energy released in a nuclear explosion. It is usually expressed in equivalent tons of TNT (the quantity of TNT required to produce a corresponding amount of energy). Low yield is generally considered to be less than 20 kilotons; low intermediate yield from 20 to 200 kilotons; intermediate yield from

200 kilotons to 1 megaton. There is no standardized term to cover yields from 1 megaton upward. (Compare fission yield, see TNT equivalent.)

Z

The symbol for atomic number.

Multiples and Submultiples	Prefixes	Symbols
10 ¹⁸	exa	E
10 ¹⁵	peta	P
10 ¹²	tera	T
10 ⁹	giga	G
10 ⁶	mega	M
10 ³	kilo	k
10 ²	hecto	h
10 ¹	deka	da
10 ⁻¹	deci	d
10 ⁻²	centi	c
10 ⁻³	milli	m
10 ⁻⁶	micro	μ
10 ⁻⁹	nano	n
10 ⁻¹²	pico	p
10 ⁻¹⁵	femto	f
10 ⁻¹⁸	atto	a

APPENDIX C

SUMMARY OF EXTERNAL REVIEW COMMENTS ON DRAFT REPORT

In order to ensure that this report meets high standards of quality and objectivity, it has been reviewed internally and externally. The internal review was performed by both the Caltech/JPL Nuclear Waste Management Review Board and appropriate members of JPL management. Following this internal review the report was modified to incorporate many of the improvements that were suggested by the internal reviewers. It should be noted that during this process, the authors were at complete liberty to either include or reject suggestions as they saw fit.

A final draft of the report was then submitted to external review by a sampling of persons or organizations who were judged by the authors to have significant insight into the current state of high level nuclear waste management. A list of the persons and organizations is presented in Table C-1. A conscious attempt was made to choose knowledgeable external reviewers who represent a broad variety of viewpoints including those of environmentalists, utilities, and pertinent state and federal officials. We have received review comments from the Department of Energy, the Nuclear Regulatory Commission, the Environmental Protection Agency, the California Energy Resources Conservation and Commission, and the Swedish Embassy. In general the comments of the reviewers are presented following this introduction. In the case of the comments from the California Energy Resources Conservation and Development Commission's Nuclear Assessments Office, only the summary comments (5 pages) are included since they chose to make extensive comments in the margin of the text. These comments are on file at JPL for interested parties.

Following this external review, the report was modified to incorporate many of the suggested improvements. Once again, it should be noted that during this final revision the authors were free to incorporate or reject comments according to their own judgment. Hence, the responsibility for the contents of this report is solely that of the authors. The authors are grateful to all of these reviewers for making helpful comments and criticisms which contributed to improving this report.

Table C-1. External Reviewers of This Report

NAME	ORGANIZATION
Mr. Phil Compton	NASA Headquarters
Dr. Joel Snow	Office of Science and Technology Policy
Mr. Chuck Guttman	Marshall Space Flight Center
Dr. William Bishop	Nuclear Regulatory Commission
Dr. William Rowe	Environmental Protection Agency
Dr. Hannes Alfven	Scripps Institution of Oceanography
Mr. Amory Lovins	Friends of the Earth, Ltd
Mr. James Helt	Dept. of Water & Power (L.A.)
Dr. James Teem*	Association of Universities for Research in Astronomy, Inc. (AURA)
Dr. John Holdren	University of California, Berkeley
Dr. Gerald Brubaker	Council on Environmental Quality
Dr. Lars Helander	Swedish Embassy
Dr. Fredrick Weinhold	Energy Office, Executive Office of the President
Dr. John Ahearne	Energy Office, Executive Office of the President
Dr. Terry Lash	Natural Resources Defense Council
Dr. Carl Kuhlman	Energy Research and Development Administration
Dr. Alex Perge	Energy Research and Development Administration
Dr. Pasternak	California Energy Resources Conservation and Development Commission
Mr. Varanini	California Energy Resources Conservation and Development Commission
Dr. Doctor	California Energy Resources Conservation and Development Commission
M. Moretti	California Energy Resources Conservation and Development Commission
Mr. Maullin	California Energy Resources Conservation and Development Commission
Mr. Golliher	California Energy Resources Conservation and Development Commission
Nuclear Assessment Office Staff	Office of Technology Assessment
Dr. Joseph Coates	Lowenstein, Newman, Reis and Axelrad
Mr. Maurice Axelrad	Nuclear Safety Association
Dr. Joe Lieberman	Tennessee Valley Authority
Mr. Bryant Brooks	Energy Research Group
Mr. Jim Muckerheid	Yankee Atomic
Mr. Alan Hansen	Duke Power
Mr. Ralph Postian	Department of Energy, ERDA, San Francisco
Mr. Harvey Bristol	San Diego Gas & Electric
Dr. Lou Bernath	Southern California Edison
Mr. Gene Cramer	Pacific Gas & Electric
Mr. Owen Davis	SMUD
Mr. Roger Powers	Los Angeles Department of Water & Power
Mr. Dan Brosnan	

* Member of Nuclear Waste Management Review Board while at Caltech



Department of Energy
Washington, D.C. 20545

OCT 23 1977

Dr. Thomas D. English
Project Manager
Nuclear Waste Management
Jet Propulsion Laboratory
California Institute of Technology
4800 Oak Grove Drive
Pasadena, CA 91103

Dear Dr. English:

Thank you for the opportunity to comment on your September 16, 1977 draft of "An Analysis of the Technical Status of High-Level Radioactive Waste and Spent Fuel Management Systems." My compliments to you and your associates on a thorough, well-organized treatment of a complex subject.

In my opinion, the description of technology and technical programs is comprehensive and presented in a balanced professional manner. The conclusions with respect to feasibility and achievability I agree with in general. However, I tend to take exception to the conclusion that scientific feasibility of deep geologic isolation in bedded salt is not proven. Viewed from the perspective of man's knowledge of the rates and mechanisms associated with significant geological changes, we can certainly say with confidence that a site can be found in a bedded salt formation where the risk to man as a consequence of the emplacement of radioactive waste would be insignificant.

Sincerely,

Alex F. Perge
Senior Technical Assistant
Division of Waste Management,
Production and Reprocessing



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 18 1977

Dr. T. D. English
Jet Propulsion Laboratory
California Institute of Technology
Pasadena, California

Dear Dr. English:

This confirms our phone conversation of the 7th. During that conversation, I made the following observations on your document 5030-90.

1. The discussion of INEL's experience in calcining liquid waste and spent fuel storage did not appear to reflect the relative status accurately.
2. The discussion of thermal effects resulting from the emplacement of waste in a repository did not include a far field consideration. (I have arranged for a copy of our limited study in this area to be sent to you.)
3. The discussion on page 6-44 regarding near surface heat-sink storage suggested technical limitations, and did not reflect that possibility that avoidance of this option may well be a political or legal (patent) question.
4. The discussion on page 7-5 regarding NRC's ownership of fuel did not include an explanatory statement about the existence of a typographical error in the rule.
5. Relative to the statement on page 7-9 regarding NRC hearings delaying operation of a repository, it is my opinion that NRC activities need not act to delay the initiation of operation of a repository. As you noted elsewhere in the report, the major delay appears to be the lack of decisive action by the Federal Government.
6. It seemed to me that the underlined emphasis on page 8-11 tends to bias the reader, before he reads the following qualifying sentences.

As I mentioned in our talk, I feel that your study is a very thorough and relatively balanced review of waste management. I hope that the

Dr. T. D. English

" - 2 -

California board makes the fullest use of it. I would appreciate receiving several additional copies of it and of your report JPL 77-59, Volumes I and II.

Please call on us for further consultations at any time.

Sincerely,



S. J. S. Parry
High-Level and Transuranic
Waste Branch
Division of Fuel Cycle
and Material Safety



UNITED STATES ENVIRONMENTAL PROTECTION AGENCY
WASHINGTON, D.C. 20460

3 NOV 1977

Dr. Thomas D. English
Project Manager, Nuclear Waste Management
Jet Propulsion Laboratory
California Institute of Technology
4800 Oak Grove Drive
Pasadena, California 91103

Dear Tom:

Thank you for the opportunity to review your draft report "An Analysis of the Technical Status of High-Level Radioactive Waste and Spent Fuel Management Systems." The reviewer found it to be a generally thorough and well-written report.

There seems to be some problem with the appendices, however. Section 7.3 mentions examining the nuclear waste programs of EPA, NRC and ERDA and refers to an examination of material from EPA, NRC and ERDA in appendices B, C, and D, but the draft report contains no appendices C and D, and has a glossary as appendix B. Appendix A is also significantly inferior in quality to the main body of the report.

Detailed comments are enclosed. I hope they prove useful to you in preparation of the final report.

Sincerely yours,

A handwritten signature in dark ink, appearing to read "W. D. Rowe".

W. D. Rowe, Ph.D.
Deputy Assistant Administrator
for Radiation Programs (AW-458)

Enclosure

C-4

NOV 9 1977

Comments on Draft JPL Report "An Analysis of the Technical Status of High-Level Radioactive Waste and Spent Fuel Management Systems"

1. This draft report provides too little discussion of disposal alternatives other than geologic isolation in bedded salt, and of containing media other than borosilicate glass. Approaches which might be given serious consideration by other countries, such as disposal in rock in the seabed, should be treated more extensively.
2. Some remark should be added to the summary with regard to the reasons for the emphasis in this report on deep geologic isolation in bedded salt and on borosilicate glass as the containing medium for high-level waste (HLW), rather than on other potential choices.
3. A statement should be added to the Introduction explaining the recent creation of the Department of Energy (DOE) and the dissolving of the Energy Research and Development Administration (ERDA) and its Office of Waste Isolation (OWI).
4. The discussion on page 1-4 should acknowledge that the Nuclear Regulatory Commission (NRC) has draft proposed site suitability criteria.
5. Environmental and safety considerations should be specifically addressed in the analysis of major components (referred to in the first complete paragraph, page 2-2), in order to indicate whether they are potential obstacles to successful utilization of the components.
6. The discussion in the Introduction of the content of the report's sections refers to the sections by Roman numerals although they are numbered in Arabic numerals, and the numbers do not correspond throughout. For instance, there is no section IX; section 8 contains the conclusions.
7. Some discussion of inconsistencies in the literature with regard to canister spacing and temperature of adjacent salt should be made to indicate the lack of decision on such design considerations. Similarly, the chronology and sequence of operations in the repository mine is ill-defined.
8. The last paragraph on page 4-1 ascribes too great a thermal power to irradiated fuel "at the time of discharge." The value given is closer to that at the time of shutdown of the nuclear chain reaction. Section 6.1.1 implies delays of a week or more before discharge.
9. The last sentence of the second paragraph on page 4-4 is misleading with regard to the importance of corrosion products; Table 4-1 to which it refers indicates that the nickel, chromium and iron make up only 1.4% of the weight of the HLW.

10. With regard to section 4.4.1, some explanation should be provided for the decision to keep the growth of nuclear capacity in the analysis the same regardless of the decision about reprocessing.
11. Some of the figures and tables should be labeled more thoroughly. One example is Figure 4-7 which has units of MTU but is entitled "Cumulative Liquid High-Level Waste...." Another example is Table 4-5 which is titled "Generation of Solidified High Level Waste Fuel Reprocessed (MT)" but does not indicate what form of material is referred to, i.e. whether it is MT of glass containing HLW or something else.
12. Quantities quoted are not consistent throughout; compare pages 4-5 and 5-1.
13. The discussion on page 5-26 on the extrapolation of leach-rate information is poor and illogical.
14. It is asserted on page 8-3 that 23 wt % waste oxides in glass is optimum, but it is asserted on page 5-28 that 28 wt % is optimum.
15. Section 5.2.3.4.3 refers to a conclusion indicating ambiently stored casks should not be subsequently stored in water. This implies that perhaps they should also not be stored in geologic media where they will be exposed to water if continued retrievability is desired.
16. Figure 5-15 gives no indication of the on-site storage of the mined rock or salt from the waste storage areas. The accompanying text (page 5-37) implies that it will not be left at the site (in that the support buildings "...will be the only visible evidence of the repository.") Some indication should be given of the plans for shipment and disposal of what is obviously more than 197,000 cubic meters of material. This is partially acknowledged in section 5.3.4.1.2.
17. On page 5-38, reference is made to sealing the repository when it is full; some estimate should be made of the year in which this will occur.
18. The argument at the bottom of page 5-38 suggests that a plurality of significant expenditures is of less concern than a single significant expenditure.
19. Some discussion should be included in Section 5.3.4 of the potential for chemical, thermal, and mechanical effects of the HLW on salt to affect the mode and duration of operation of a repository. The text, particularly that at the bottom of page 5-45, implies that the repository design is to be such that it can stand open with minimal maintenance for decades. The required behavior of the salt for this mode of operation should be considered in view of the potential effects of the HLW on the salt behavior.

20. With regard to scenario (2) in section 5.3.4.1.5, it is not clear that displacement upward of the disposal horizon is necessary if a combination of adequate water supply and sufficient fault action occurs.

21. Section 6.1 contains a wealth of unnecessary detail.

22. The reason "Because of the negligible particulate activity available for release, ..." given at the bottom of page 6-28, seems incorrect considering the source is impacted spent fuel. The presence of media and systems which will retain the material appears to be a better argument.

23. Some of the design features for spent fuel disposal appear unnecessarily costly compared to those for HLW from reprocessing. An example is the canister wall thickness of 3 cm stainless steel for the spent fuel (page 6-49) compared to 1.3 cm for the HLW from reprocessing. Similarly, the need for the repository vault to be 6.4 meters high to accommodate placing canisters only 4 meters long is not apparent. Also, the difference in canister diameters would be unlikely to influence the mine shaft diameter appreciably.

24. Section 6.4.4.3 contains a comparison of the spent fuel activity at different times to the radium-226 activity in the original uranium ore; this comparison in curies is incorrect because the ore commonly contains not just uranium and radium but all the other radioactive products of the decay of uranium.

25. Section 7.3 refers to material in appendixes B, C and D which was not included in the draft report.

26. Section 8.4 discusses shipment by rail, and then cask requirements. To avoid misunderstanding, the discussion of cask requirements should distinguish between rail casks and truck casks.

27. Some reference should be provided for the assertion on page A-7 that "...some detectable quantity of radioactivity from every phase of the fuel cycle will appear in air and water", if it applies to a sealed repository functioning as intended.

28. In the statement on page A-8 "At present, the radioactivity of this military waste exceeds that of the accumulated commercial waste.", insert "reprocessed" before "commercial." The amount of radioactive waste contained in stored spent fuel is appreciable.

29. The example using krypton-85 (page A-10) can be faulted for a number of reasons. It is most easily corrected by discussing instead the aspect of the krypton-85 in the air as source of beta radiation irradiating the skin. See NCRP Report No. 44.

30. The statements on page A-10 with respect to regulation of waste disposal sites and 10 CFR Part 20 may be misleading, in that, although EPA has not yet proposed an environmental radiological protection standard for management of HLW it is unlikely that the standard when promulgated will permit radiation doses as large as those permitted by 10 CFR Part 20.

31. EPA has authority to set generally applicable environmental protection standards and has used it to promulgate a standard applicable to the operations of the uranium fuel cycle (excluding mining, transportation, and waste). A copy of the standard, 40 CFR Part 190, is enclosed for reference in revising the material at the top of page A-11.

32. On page A-11, the reference to accidental exposure of human beings should be revised so that it does not indicate that the experiences at Hiroshima were due to an accident.

33. On page A-12, delete "(presumably per year)."

34. The parenthetical remark at the bottom of page A-14 is seriously in error, because only a minor fraction of the mined uranium is contained in the spent fuel (the rest is enrichment plant tails), because efficient commercial recycling of fission fuels is yet to be satisfactorily demonstrated and is unlikely to achieve such efficiency, and because some of the uranium is converted into isotopes of uranium and plutonium which are relatively useless in fission reactors.

35. It is not considered likely that repository closure will include intentional collapse of the overlying rock as indicated on page A-16.

THURSDAY, JANUARY 13, 1977

PART VII



**ENVIRONMENTAL
PROTECTION
AGENCY**



**RADIATION PROTECTION
PROGRAMS**

**Environmental Radiation Protection
Standards for
Nuclear Power Operations**

Title 40—Protection of Environment
CHAPTER I—ENVIRONMENTAL
PROTECTION AGENCY

SUBCHAPTER F—RADIATION PROTECTION
PROGRAMS
 [FRL 659-6]

PART 190—ENVIRONMENTAL RADIATION
PROTECTION STANDARDS FOR NU-
CLEAR POWER OPERATIONS

On May 10, 1974, the Environmental Protection Agency (EPA) published an advance notice of intent to propose environmental radiation protection standards for the uranium fuel cycle (39 FR 16906) and invited public participation. On May 29, 1975, EPA proposed regulations setting forth such standards (40 FR 23420) pursuant to the Atomic Energy Act, as amended, and Reorganization Plan No. 3 of 1970 (35 FR 15623). Numerous written comments were received, and a public hearing was held on March 8-10, 1976 (41 FR 1124 and 41 FR 5349).¹

These regulations setting forth environmental radiation standards are hereby promulgated in final form. The standards specify the levels below which normal operations of the uranium fuel cycle are determined to be environmentally acceptable. A number of changes have been made in the proposed regulations in response to comments received. These changes modify and clarify the areas of applicability of the standards and their effective dates, and expand the conditions under which variances may be granted. The numerical levels of the standards have been retained as proposed.

The Agency has benefited from extensive public participation during the course of the development of these regulations. Sixteen comment letters were received in response to the Agency's May 10, 1974, notice of intent to propose standards, and 82 comment letters following the publication of proposed regu-

lations on May 29, 1975. Letters were received from a broad cross-section of representatives of the general public, the industry, professional groups, the States, and Federal agencies. In addition, 17 parties participated in three days of public hearings and, in many cases, submitted extensive additional written testimony. In all, the contributed record comprises over 3500 pages. Comment letters, a transcript of the public hearing, and all submitted testimony are available for viewing and copying in the Agency's Public Information Reference Unit, Room 2922, U.S. Environmental Protection Agency, 401 M Street SW, Washington, D.C. 20460. The Agency has considered all of this record in reaching its conclusions for these final regulations.

At the time these standards were proposed, EPA released a Draft Environmental Statement and solicited public comments. A Final Environmental Statement is being made available concurrently with the promulgation of these standards. This statement contains the comments received on both the proposed standards and the draft statement, and EPA's response to these comments. Single copies of the Final Environmental Statement and an additional document containing EPA's detailed responses to testimony received in connection with the public hearing are available from the Director, Criteria and Standards Division (AW-460), Office of Radiation Programs, Environmental Protection Agency, Washington, D.C. 20460. Persons interested in a summary discussion of the background, rationale, interpretation, and significance of these standards should consult the notice proposing these regulations and, for greater detail, the Final Environmental Statement.

MAJOR ISSUES RAISED DURING REVIEW

Three major issues were raised by commenters. These were: (1) concern that procedures for implementation of the standards would be unnecessarily conservative or costly, (2) disagreement over the need for and cost-effectiveness of control of environmental releases of krypton-85 and other long-lived radionuclides, and (3) disagreement over the form of the relationship between effects on health and radiation dose assumed in deriving these standards.

A large number of commenters expressed the view that implementation would lead to more restrictive control of effluents than intended due to the use of unnecessarily conservative models for source terms, control capability, and environmental transport, and due to requirements for unreasonably large margins between normal operating levels and the standards, especially at sites containing a number of facilities. The authority to regulate fuel cycle facilities under these standards resides in the Nuclear Regulatory Commission (NRC), or, in some cases, the States, under agreements with NRC. The standards have been expressed in terms of the dose to members of the public, rather than to hypothetical receptors, in order to encourage the use of realistic models by the regulatory agency. In addition, the

Agency has made its intent regarding realistic implementation clear, as, for example, in the discussion of these matters in the Final Environmental Statement and will continue to do so if necessary as implementation proceeds, to assure that unnecessary conservatism does not occur.

In this regard, the NRC has recently issued a revised set of regulatory guides for light-water-cooled reactors which implement their announced intent to use the most realistic models available when adequate experimental data exist to permit a prudent and scientific determination. These models are intended for use in implementing the recently-issued Appendix I to 10 CFR Part 50, which defines design and operating criteria for single reactor units. EPA has examined Appendix I and the accompanying regulatory guides and agrees that they provide the basis for realistic implementation of these standards for single reactor units. The existence of these requirements, coupled with the realization that most existing reactor licenses are for no more than one or two units on a site, makes it unnecessary, in the Agency's judgment, to reexamine the license conditions of these licensees for compatibility with these standards, unless the nearest neighboring site covered by this standard is within ten miles. In these latter cases small adjustments may be necessary. However, in the vast majority of situations, the sum of all reasonably postulated contributions from sources other than the immediate site will be small compared to these standards and should be ignored in assessing compliance. It would not be reasonable to attempt to incorporate into compliance assessment doses which are small fractions of the uncertainties associated with the determination of doses from the primary source of exposure. The Agency has also concluded that, except under highly improbable circumstances, conformance to these criteria should provide reasonable assurance of compliance with these standards for up to five units on a site. This conclusion is based, among other considerations upon realistic consideration of anticipated site sizes and the relative location of individual units, as well as the stochastic nature of effluent releases.

A number of commenters including the NRC, also noted that shutdown of nuclear facilities for minor deviations from the standards would not be reasonable. The Agency agrees, and notes that the use of such an extreme measure is not required under present compliance procedures for licenses issued pursuant to the Atomic Energy Act, and that these regulations do not add such a requirement. A graded scale of action is an appropriate regulatory response for achieving conformance. This may include, for example, requirements for corrective actions, appropriate penalties, and, in extreme cases, cessation of operations. The Agency is confident that the NRC will implement these standards in such a reasonable manner.

Some commenters expressed the view that it was not feasible to monitor con-

¹ In this connection the Agency received requests on behalf of Allied-General Nuclear Services (AGNS) on October 4 and December 2, 1976, for a supplemental hearing on certain aspects of this rulemaking, on the grounds that the Agency is, in part, relying upon information acquired subsequent to the public hearing which, in the view of AGNS, would be an essential basis for the rulemaking but is erroneous. The Agency has reviewed the materials submitted in support of this request and concluded that they would not provide a sufficient basis for altering its conclusions. A response to new matters addressed by this material has been appended to the Agency's commentary on testimony received in connection with the public hearing on these standards. In addition it is noted that the Agency has previously (40 FR 23420) made public its intent " . . . to maintain a continuing review of the appropriateness of these environmental standards . . . and to revise them, if necessary on the basis of information that develops in the interval." In view of the above, the Agency has concluded that it is neither necessary nor appropriate to grant now the additional public hearing requested. We will, of course, welcome the submission of additional factual data on the matters concerned as it becomes available.

RULES AND REGULATIONS

2859

formance with these standards through the use of environmental measurements. The Agency agrees that routine monitoring based exclusively upon environmental measurements would not be a reasonable means for assuring conformance and the regulations do not contain such a requirement. Environmental objectives are generally best achieved through controls exercised at the source. For this reason effluent monitoring is generally preferable and such measurements, when combined with regulatory models for environmental transport, would provide quite adequate demonstration of conformance with the standards for the vast majority of situations, based upon existing experience. However, since varying degrees of conservatism and uncertainty exist in all environmental models, the Agency believes it will often be appropriate to supplement effluent monitoring with confirming environmental measurements, as is now the regulatory practice. In the case of light water reactors, models and monitoring requirements for demonstrating conformance with Appendix I of 10 CFR Part 50 are generally adequate for demonstrating conformance with these standards. Similar models and measurements would, in general, be appropriate for most other types of facilities.

In the special case of possible wind-blown effluents from mill tailings, the existence of operational measures (e.g., temporary or permanent stabilization) should normally be the criterion used for verifying compliance, in lieu of effluent and environmental monitoring, because of the difficulty associated with such measurements. It should be noted that doses resulting from exposure to radon and its daughters, which are discharged from a mill site (or result from material which has been discharged), are excluded, but that gamma radiation crossing site boundaries from any on-site source is covered.

In situations where members of the public are actually exposed, these standards, in effect, preempt those regulations which are based upon the Federal Radiation Protection Guides (25 FR 4402) insofar as exposure of the public is due to operations defined to be included in the uranium fuel cycle. For example, the dose limits in 10 CFR Part 20 would not be the limiting consideration regarding exposure of members of the public as a result of uranium fuel cycle operations. These standards do not, however, replace application of the Radiation Protection Guides to the regulation of sources not included within the scope of the uranium fuel cycle. Finally, the graded scale of actions established in 1961 (26 FR 9057) for use in implementing the Radiation Protection Guides do not apply to implementation of these standards, but would remain in effect for implementation of radiation protection guides for other radiation sources.

Several commenters expressed the view that a requirement for control of the unrestricted release of krypton to the environment from fuel cycle operations

was: (a) beyond the jurisdiction of EPA, (b) unreasonably costly, (c) not achievable by 1983, the proposed implementation date (or, in the view of some commenters, was achievable prior to 1983), or (d) not a reasonable requirement of domestic industry until international agreements are achieved to restrict emissions from foreign sources.

The Agency has concluded that its jurisdiction is clear. Reorganization Plan No. 3 of 1970 specifically transferred to EPA from the Atomic Energy Commission the authority to establish standards for "quantities of radioactive materials in the environment" and attaches no conditions to this authority except a requirement that the standards apply outside the boundaries of licensees.

EPA has carefully reexamined the costs of control systems for krypton and has concluded that a substantial portion of the additional costs presented at the public hearings is correct. This analysis is reviewed in the Final Environmental Statement. However, in spite of these increased costs, the installation of controls for krypton-85 is believed to be justified by the public health benefits achievable. In today's dollars, the cost per unit radiation dose reduction at future reprocessing facilities will be \$50-\$75 per man-rem for whole body doses, and considerably less than this for doses to other organs. These values are more than an order of magnitude lower than limiting costs now specified in regulations governing the licensing of individual nuclear power reactors. It is recognized that the cost of retrofitting one facility which is expected to be in operation before 1983 will involve greater costs, and the regulatory agency is encouraged to explore means to minimize costs to this facility in its implementation of the standard for this pilot case.

Regarding the achievability of control over the release of krypton-85 to the environment by 1983, it is noted that this or similar control technology is already being offered commercially for nuclear reactors and fuel reprocessing facilities, and is currently being installed, or is on order, at several U.S. reactors and at a foreign fuel reprocessing facility by U.S. suppliers. The Agency, therefore, believes that 1983 is an achievable implementation date. However, a more accelerated schedule is not considered justified, in view of the small amount of reprocessing that will occur before that date and the present lack of operating experience with krypton controls.

Finally, we have examined arguments concerning the need for international agreement prior to the establishment of standards and do not find them persuasive. EPA fully supports the development of international agreements, and is presently participating in the development of international guidance for control of radioactive effluents from the fuel cycle under the auspices of the International Atomic Energy Agency. A number of countries are already committed to or are in the process of committing themselves to control of krypton releases. The Agency supports this trend and has con-

cluded that the control of U.S. releases of krypton-85 is warranted on the basis of reducing its potential worldwide public health impact. In initiating a requirement for this control, the United States fulfills its responsibility, as the world's largest user of nuclear power, to provide leadership in this matter.

A number of commenters suggested that the proposed regulations should be amended to include standards for carbon-14 and, in some cases, other long-lived radionuclides. The Agency has studies of sources and controls for these materials underway and anticipates that proposals for appropriate environmental standards for carbon-14 can be made shortly, with consideration of proposals for other materials following at a later date. However, the knowledge base is not yet sufficient to permit incorporation into these standards now.

Comments were received reflecting many points of view on health effects issues. One group agreed with the Agency's primary reliance on risk estimates provided by the recent report to EPA of the National Academy of Sciences ("The Effects on Populations of Exposure to Low Levels of Ionization," Report of the Advisory Committee on the Biological Effects of Ionizing Radiation, NAS-NRC, 1972). These estimates are primarily based upon a linear interpolation between existing data on human populations and the assumption of no effects at zero dose. Another group believed this model is not sufficiently conservative to adequately protect public health, based upon several investigators' hypotheses concerning the shape of the dose-effect relationship at low doses. A third group believed these estimates to be too conservative at low doses and low dose-rates. Frequent reference was made by the third group to a report of the National Council on Radiation Protection and Measurements (Report No. 43) which implies that radiation standards should not be based upon numerical estimates of health effects, and a recent report of the Nuclear Regulatory Commission (NUREG-75/014) which presents, in addition to risk estimates based upon the National Academy of Sciences report, some lower risk estimates based upon a belief that dose-rate dependent phenomena exist for low linear energy transfer radiation (gamma rays and beta particles) which reduce the carcinogenic effect of radiation to levels lower than those predicted by the linear model. The Agency has examined the evidence for each of the above views and concluded that, while each may have validity under various assumptions or for various specific situations, the weight of currently available scientific evidence supports the continued use of a linear, nonthreshold model for deriving standards to protect public health.

Changes Made in the Proposed Regulations

A number of changes have been made in response to comments received on the proposed regulations. The following describes and provides the reasons for each of these changes.

1. Paragraph 190.02(b) has been changed to delete transportation as an operation covered by these standards and to specifically exclude waste disposal sites, which were previously not mentioned. The Agency is addressing the development of criteria and standards for management of radioactive wastes as a separate matter, as mentioned in the notice proposing these standards.

A number of commenters, including the NRC and the Department of Transportation, pointed out the difficulty of implementing these standards for transportation activities, particularly noting the problems near nuclear facilities. In such cases an apportionment of the dose limits would appear to be necessary in order to avoid unreasonably extensive monitoring requirements for members of the public. Since studies by both EPA and NRC show that most transportation-related doses are expected to remain at small fractions of these standards in any case, the implementation difficulty does not appear to warrant their inclusion in these standards limiting doses to individuals from uranium fuel cycle operations. The Agency will instead address this matter under its broad authority inherited from the former Federal Radiation Council, through the development of more general guidance to all Federal agencies concerning radiation exposure arising from the transportation of all types of radioactive materials, not just those from the uranium fuel cycle.

2. Paragraph 190.02(d) is changed to reflect the definition of "site" implied by Reorganization Plan No. 3 of 1970.

3. Paragraph 190.02(f) is changed by adding the word "spontaneously" to reflect the Agency's original intent.

4. Paragraph 190.02(g) is deleted and subsequent paragraphs in Section 190.02 are renumbered. This paragraph defined uranium ore as ore containing 0.05% or more uranium by weight. As pointed out by one commenter, it is not desirable to exclude ores containing less than this quantity of uranium, since future demand for ore may make the use of such ores economically feasible.

5. Section 190.11 has been broadened to permit a greater degree of discretion to the regulatory agency to develop and apply conditions for the granting of variances. As pointed out by a number of commenters, it is not reasonable to predicate the justification for variances solely on public need for orderly delivery of power. For example, a facility may have installed a control system which, in spite of good faith performance on the part of the supplier and the user, may fail to achieve operational capability on a timely basis, or, once installed may experience operational failure at some time, yet operation of the facility may not be essential to the "orderly delivery of electrical power." In addition, some portions of this standard are predicated upon the use of waste treatment systems not yet in general commercial use. Although in no case should operation continue if safety is compromised, it may easily be that excursions above these

standards would occur in such cases to a degree that the added risk to the general public is small and the environmental effect is acceptable in comparison to the economic penalty that would be associated with cessation of operation or the anticipated public health and environmental impact of available alternative sources of power. For this reason, the variance provision has been broadened so that the regulatory agency may, if it deems it to be in the public interest, grant a variance in such situations. It should be noted, however, that the variance provision applies only to temporary and unusual situations. It is expected that continued operation under the variance provision will be predicated upon an approved plan to achieve compliance in an expeditious fashion, that is, in as short a time as is reasonably achievable.

The requirement for public documentation of variances has been clarified and extended to apply to this broadened provision. EPA will not review individual variances or compliance plans, which will be made public in accordance with the provisions of paragraph 190.11(b), but will maintain a general overview through periodic review of the use of this Section.

6. Section 190.12(a) has been changed to provide that the effective date for the standards limiting doses to individuals shall be December 1, 1979, for all operations except the milling of uranium ore, for which the effective date shall be December 1, 1980.

The NRC has carefully examined its existing programs for implementation of Appendix I at light-water-cooled reactors, and the feasibility of integrating implementation of these standards into that on-going process, as well as, in parallel, implementing these standards at other types of fuel cycle facilities through development and promulgation of new regulatory guides and individual license conditions. Finally, there are matters regarding reactors which will require generic treatment, such as the conditions required for compliance when there are multiple units on single sites. It is the conclusion of the NRC, and the Agency concurs, that the originally proposed two-year implementation period is insufficient and that three years will be required to complete this process. The NRC review of these matters regarding implementation has revealed that the case of mills is unique, since better information is required concerning a number of alternatives for stabilization of tailings—both as to their relative merit and the degree of periodic maintenance required. On June 3, 1976, the NRC published (41 FR 22430) a notice of intent to prepare a generic environmental statement on uranium milling operations. This effort will be completed in approximately two years, and includes field measurements with participation of both EPA and NRC personnel. In addition, the NRC issued proposed new effluent reporting requirements at mills on November 17, 1975 (40 FR 53230). In view of the above considerations it is the jointly agreed upon conclusion of

the Agency and NRC that a four-year implementation period is required at mills, rather than the three years provided for all other fuel cycle operations.

7. Section 190.12(b) has been changed to clarify the Agency's original intent that the standards specified in paragraph 190.10(b) apply to radioactive materials produced after the effective date.

The Agency anticipates that promulgation of these standards will serve, in addition to providing for necessary protection of public health, to alleviate some of the uncertainties associated with the design of environmental controls for fuel cycle facilities, and the consequent economic penalties, through stabilizing and providing direction to the process of development of standards and regulations. The economic and inflationary impacts of these regulations have been evaluated in accordance with Executive Order 11821 and it has been determined that an Inflation Impact Statement is not required. (The estimated annual cost of additional effluent controls required by these regulations is in no case greater than ten to twenty million dollars, which is significantly less than the one-hundred million dollar annual cost cut-off established as the minimum for which an Inflation Impact Statement is required.)

Notice is hereby given that pursuant to the Atomic Energy Act of 1954, as amended, and Reorganization Plan No. 3 of 1970 Title 40, Chapter I, of the Code of Federal Regulations is amended by adding a new Subchapter F and Part 190 as set forth below.

Dated: December 28, 1976

RUSSELL E. TRAIN,
Administrator.

A new Subchapter F, consisting of Part 190, is added to 40 CFR Chapter I as follows:

SUBCHAPTER F—RADIATION PROTECTION PROGRAMS

PART 190—ENVIRONMENTAL RADIATION PROTECTION STANDARDS FOR NUCLEAR POWER OPERATIONS

Subpart A—General Provisions

Sec

190.01 Applicability.

190.02 Definitions.

Subpart B—Environmental Standards for the Uranium Fuel Cycle

190.10 Standards for normal operations.

190.11 Variances for unusual operations

190.12 Effective date.

AUTHORITY: Atomic Energy Act of 1954, as amended; Reorganization Plan No. 3, of 1970

Subpart A—General Provisions

§ 190.01 Applicability.

The provisions of this Part apply to radiation doses received by members of the public in the general environment and to radioactive materials introduced into the general environment as the result of operations which are part of a nuclear fuel cycle.

§ 190.02 Definitions.

(a) "Nuclear fuel cycle" means the operations defined to be associated with the

RULES AND REGULATIONS

2861

production of electrical power for public use by any fuel cycle through utilization of nuclear energy

(b) "Uranium fuel cycle" means the operations of milling of uranium ore, chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electrical power for public use utilizing nuclear energy, but excludes mining operations, operations at waste disposal sites, transportation of any radioactive material in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle.

(c) "General environment" means the total terrestrial, atmospheric and aquatic environments outside sites upon which any operation which is part of a nuclear fuel cycle is conducted.

(d) "Site" means the area contained within the boundary of a location under the control of persons possessing or using radioactive material on which is conducted one or more operations covered by this Part

(e) "Radiation" means any or all of the following: alpha, beta, gamma, or X-rays; neutrons; and high-energy electrons, protons, or other atomic particles; but not sound or radio waves, nor visible, infrared, or ultraviolet light

(f) "Radioactive material" means any material which spontaneously emits radiation

(g) "Curie" (Ci) means that quantity of radioactive material producing 37 billion nuclear transformations per second. (One millicurie (mCi) = 0.001 Ci.)

(h) "Dose equivalent" means the product of absorbed dose and appropriate factors to account for differences in bio-

logical effectiveness due to the quality of radiation and its spatial distribution in the body. The unit of dose equivalent is the "rem." (One millirem (mrem) = 0.001 rem.)

(i) "Organ" means any human organ exclusive of the dermis, the epidermis, or the cornea.

(j) "Gigawatt-year" refers to the quantity of electrical energy produced at the busbar of a generating station. A gigawatt is equal to one billion watts. A gigawatt-year is equivalent to the amount of energy output represented by an average electric power level of one gigawatt sustained for one year.

(k) "Member of the public" means any individual that can receive a radiation dose in the general environment, whether he may or may not also be exposed to radiation in an occupation associated with a nuclear fuel cycle. However, an individual is not considered a member of the public during any period in which he is engaged in carrying out any operation which is part of a nuclear fuel cycle

(l) "Regulatory agency" means the government agency responsible for issuing regulations governing the use of sources of radiation or radioactive materials or emissions therefrom and carrying out inspection and enforcement activities to assure compliance with such regulations.

Subpart B—Environmental Standards for the Uranium Fuel Cycle

§ 190.10 Standards for normal operations.

Operations covered by this Subpart shall be conducted in such a manner as to provide reasonable assurance that:

(a) The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any

member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations.

(b) The total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle, contains less than 50,000 curies of krypton-85, 5 millicuries of iodine-129, and 0.5 millicuries combined of plutonium-239 and other alpha-emitting transuranic radionuclides with half-lives greater than one year.

§ 190.11 Variances for unusual operations.

The standards specified in § 190.10 may be exceeded if:

(a) The regulatory agency has granted a variance based upon its determination that a temporary and unusual operating condition exists and continued operation is in the public interest, and

(b) Information is promptly made a matter of public record delineating the nature of unusual operating conditions, the degree to which this operation is expected to result in levels in excess of the standards, the basis of the variance, and the schedule for achieving conformance with the standards.

§ 190.12 Effective date.

(a) The standards in § 190.10(a) shall be effective December 1, 1979, except that for doses arising from operations associated with the milling of uranium ore the effective date shall be December 1, 1980.

(b) The standards in § 190.10(b) shall be effective December 1, 1979, except that the standards for krypton-85 and iodine-129 shall be effective January 1, 1983, for any such radioactive materials generated by the fission process after these dates

[FR Doc. 77-399 Filed 1-12-77; 8:45 am]

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ORIGINAL PAGE IS POOR

**ENERGY RESOURCES CONSERVATION
AND DEVELOPMENT COMMISSION**1111 HOWE AVENUE
SACRAMENTO, CALIFORNIA 95825

(916) 322-6925

October 18, 1977

Dr. Thomas English
Jet Propulsion Laboratory
California Institute of Technology
Pasadena, California

Dear Tom:

The Safety and Compliance Office (SCO) Staff have performed a cursory review of your report entitled "An Analysis of the Technical Status of High-Level Radioactive Waste and Spent Fuel Management Systems," dated September 16, 1977. The staff have no substantive comments to offer.

The evaluation of the health and safety aspects of the back-end of any fuel cycle is a complex problem. The problem is compounded in the nuclear industry because of the extreme sensitivity associated with the risk to the public health and safety due to long-lived radio-nuclides. It is anticipated that your report, and in particular sections 4 through 7, will be used by the Nuclear Assessments Office as source material in evaluating probabilities of potential accidents, and the consequences of these potential accidents. As the SCO staff understands, the findings from the report will be the bases for the NAO's brief to the Commission on whether the United States through its authorized agencies has approved and there exists a demonstrated technology or means for the disposal of high-level nuclear (commercial) wastes.

For the above reasons we believe that your report presents pertinent information that should be helpful in the Commission's deliberations.

Sincerely,

*Kenneth Golliher*Kenneth Golliher
Nuclear Engineer

KG:rp

cc: Frank Hahn
Larry Welsh
Darrel Woo
Gary Simon

OCT 24 1977

**ENERGY RESOURCES CONSERVATION
AND DEVELOPMENT COMMISSION**

1111 HOWE AVENUE

SACRAMENTO, CALIFORNIA 95825

(916) 322-2038

OCT 13 1977



October 7, 1977

Dr. Thomas English
Jet Propulsion Laboratory
4800 Oak Grove Drive
Pasadena, CA 91103

Dear Tom:

Enclosed are the comments of the Nuclear Assessments Office on the draft waste analysis report produced under our contract. Other comments may come to you separately from the Siting Division, the Energy Assessments Division, or the Commissioners. We have not included the comments of the other divisions at this time in order to reduce the time lag in undertaking revisions.

The basic problem with the report is that except in minor respects there is no insightful analysis -- there is little evaluation of data and experimentation, no questioning of underlying assumptions or approaches, and incorporation of conclusions of other reports without an independent check on the validity of those conclusions. In the section on spent fuel storage, major parts of the discussion and analysis are lifted wholesale from the G.E. Morris Consolidated Safety Analysis Report with little attempt to evaluate this information critically or seek other views.

This type of an approach, therefore, misses the whole point of what a "technical assessment" is intended to be. We believed a more penetrating analysis was well within the capabilities of JPL, considering the level of effort devoted to the project, and still maintain that belief. What a technical assessment requires is not merely a recitation of statements from two dozen references but thought as to what those statements mean, whether they are adequate, where the holes might be, what is left unresolved, whether experiments have resolved what the authors say is resolved, whether a report's conclusions are supported by its contents, mechanisms which might have been overlooked, and so forth. Our general feeling about your report is therefore one of disappointment.

We have had many discussions, including the one of May 27 where we expressed the same concern that too little evaluation was being done. We are appreciative of the additional details in this version of the report but had expected more from the effort undertaken.

The problem of too little evaluation comes up most clearly when a statement of the form "on the basis of an extensive review at (blank) these problems are considered resolved." Considered by whom to be resolved? Who did the

Dr. Thomas English
 October 7, 1977
 Page 2

OCT 13 1977

review? Was the review adequate? How was this established? What have others concluded? Overall, generalizations or conclusions of this type just do not appear to be justified or supported by the data in the draft report. It would be much more accurate to say "on the basis of an extensive review, the (blank) national lab reported their conclusions that the problem was resolved; JPL has not independently evaluated this work," if in fact that is the case.

The report also has some problems in overall approach. The evaluative framework, as to what is and is not included in each of the three stages, is not consistent throughout. The approach in Section 3 (pp. 3-2, 3-3) runs into inconsistencies on p. 3-12 and especially p. 6-44. Some re-thinking is required. Second, a very narrow (and, according to Carl Kuhlman, inadequate) definition of HLW is used which leads to some logic problems. Bullard's Appendix A recognizes the definitional problem and avoids the problem of the rest of the text.

In details, the report needs more documentation and elaboration. The lack of page references is annoying and, as Charles Miller and I have discussed, should be corrected.

There seem to be two sections missing which were included in the August draft and still listed on p. 6-42, "The RSSF Syndrome" and "ERDA Contradictions." We would hope to see these restored.

There are also several sweeping statements in the report which do not seem well backed-up, e.g.:

p. 5-38 In reference to Project Salt Vault, "...the events that followed indicate that the decision was scientifically and politically premature."

p. 5-50 "...any models that have been constructed are extremely simplistic, and cannot be used to make credible predictions."

While these statements may be true, as they are in the report they leap out at the reader starkly with little accompanying justification. The two statements quoted above will likely be very controversial. Your position must be fortified and, if it cannot, then modified.

The bulk of our comments are presented as marginalia to the text and they cover everything from minute details to broad commentary. Certain other general comments are attached as a separate piece.

If you believe a conference on these comments is in order, we would be pleased to oblige. We believe our comments will lead to a much-improved document and would hope you agree.

Sincerely yours,

GARY SIMON
 Nuclear Assessments Office

cc: Rhoads Stephenson
 Jim Walker
 Lloyd Forrest

GENERAL COMMENTS ON "AN ANALYSIS OF THE TECHNICAL
STATUS OF HIGH-LEVEL RADIOACTIVE WASTE AND SPENT FUEL
MANAGEMENT SYSTEMS," JPL 5030-90

Page 1-4, Paragraph 2

There are two assumptions in this paragraph:

1. Scientific investigations will produce a feasible answer to waste disposal.
2. The "hydrogeological" uncertainties are the major problems with this segment of the fuel cycle.

The fact that the time limitations are very real may prove to be a significant problem in seeking an answer through scientific investigations, and there are other uncertainties which are as real as the "hydrogeological" uncertainties. For example, site selection may turn out to be an increasingly real problem in the face of state opposition. Another problem is long range predictability in the geologic environment.

Section 5.3, Deep Geologic Isolation in Bedded Salt

The major concern in this section is with the hydrology of the site and its possible connection with transport of wastes from the repository. Nowhere does this section deal with problems of stability (tectonic) and seismic hazard.

Page 5-40, Paragraph 3

This paragraph gives a simplistic view of the virtues of a bedded salt deposit for the disposal of nuclear waste. Indeed it speaks well for such a soluble substance to be sitting around, i.e., the lack of circulating groundwater, which is considered the chief transporting agent for radionuclides. However, more should be learned about the nature of the cracks that apparently "heal" so easily. For example, one area of future research might include the development of a salt dome from a bedded salt deposit. At what stage do the cracks become large enough to facilitate movement of the salt upward?

Page 5-44, "Weakest Links"

There is no mention of the full scope of the study undertaken, and therefore to discuss only the weakest links is not as enlightening as it may seem. What are the actual strengths in the site selection and evaluation criteria as determined by OWI? It does not speak well that the "weakest" aspects of the process are all related to the hydrology. If we recognize salt as a good repository medium due to its dryness, it is particularly bad that we cannot characterize the site hydrology well enough to insure that state. As an analogy, it would be like promoting a restaurant based on the quality of its chef and not being able to insure the chef would be there next month ...would you still want to go next month?

-2-

Point (3) "extrapolation..." seems unrealistic. I am not sure geologists have the capability to predict anything over one million years.

Page 5-46, Project Salt Vault

The information regarding Project Salt Vault should be made much clearer. It seems that a project which fails must have in it some basic misconception in its assumptions. To then say that the results of that project are still valuable sources of information could be a mistake. Someone must pour through the proceedings and try to discover why it failed before the information generated can be labelled reliable.

Page 5-48, Paragraph 3

I disagree that "analysis of potential effects...is very much site specific." Certainly to the extent that each site is slightly different chemically this is true; however, to put off experimentation seems short-sighted. Experimentation with salt of varying compositions at this state could build a factual base for aiding in site selection.

Page 5-49, Section 5.3.4.1.5

Given the importance of this type of prediction, I think the fact that the "data base...is at present inadequate to model the present and future ground-water regimes" is a very significant observation. It seems from this statement that really the data base should be used as a major "weak link" in discussing geologic storage.

Page 5-50, Section 5.3.4.2

On page 1-4 it was stated that the hydrogeological uncertainties could be solved by scientific investigations now existing or in the planning stages. It seems from this statement that this is the only large issue left. However, in section 5.3.4.2 we learn that engineering achievability, a major part of the evaluation attempted by this study, is still non-existent due to the lack of a reliable data base. I would then conclude that the actual engineering of the repository might be another "weakest link."

Section 6, Spent Fuel Storage and Disposal

1. Parts 6.1.1 - 6.1.2 are rewritten from the San Onofre FSAR (Ref. 6-1). This very narrow discussion of refueling at one particular type of reactor does not contribute significantly to an analysis of spent fuel handling.
2. The following part (6.1.3) is far more relevant but needs expansion. Sources should be given for the figures on length of pool storage (we were given different ones in the hearings).

-3-

The discussion of spent fuel storage capacity demand and projections is very thin. There is no comparison of ERDA and AIF projections, and no explanation of the potential significance of encroachment on full-core reserve capacity or reload discharge capacity.

3. An overall shortcoming here and throughout most of the section is the inadequate documentation. Even if just one document is used, page numbers should be specified.
4. The description of independent spent fuel pool storage consists almost entirely of a summary of selected portions of the G.E. Morris CSAR (Ref. 6-15). This does not constitute an analysis of generic concerns associated with independent pool storage.
5. One significant piece of data in the CSAR was entirely omitted. The "cask tilt" accident with liner penetration should have been examined for its possible significance for accident analysis. Was this accident anticipated in any fault-tree or other type of analysis? At which points did water losses exceed the capacity of pumps to maintain water level or get rid of excess water? Ignoring this incident while using much of the other information in the CSAR is extremely inconsistent.
6. The section on SURFF is the best part of the chapter in terms of independent analysis. There was some over-reliance on the Technical Alternatives Document, and more information on the ongoing research programs and the Canadian experience would have been useful. The "Issues" section was very good, except that two of the four issues (the RSSF Syndrome and ERDA Contradictions) were apparently dropped. Will these be put back in?

SWEDISH EMBASSY
OFFICE OF THE SCIENTIFIC COUNSELOR

ADDRESS	SWEDISH EMBASSY 600 NEW HAMPSHIRE AVE N W WASHINGTON D C 20037 U S A
TELEPHONE	(202) 965-4100
TELEGRAM	SVENSK WASHINGTON
TELEX	RCA 248347 WU 89.2724

October 11, 1977

OCT 13 1977

Dr. Thomas D. English
Project Manager
Nuclear Waste Management
Jet Propulsion Laboratory
4800 Oak Grove Drive
Pasadena, Cal. 91103

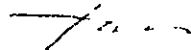
Dear Tom,

I very much appreciate your sending me a copy of your HLW-management report to the California Energy Commission. You are really doing a very good job in collecting available information in this area in a nutshell; though a very big and goodtasting nut.

You will find enclosed summaries in English of the reports that have been published so far by the Swedish Nuclear Utilities' joint nuclear fuel safety project; a project that has been formed in order to try to provide the necessary information to fulfil the Swedish fuel cycle law enacted last spring (very similar to the California fuel cycle laws of June 1976). I will be happy to send you the full reports (in Swedish) if you find that any are of special interest to you.

Please give me a call if you are going to Washington again, and we might get together and talk for a while, without me having to rush like last time.

Sincerely yours,


Lars-Inge Helander

LIH/EJo
Encls.

APPENDIX D

Assembly Bill No. 2822

CHAPTER 196

An act to add Section 25524.2 to the Public Resources Code, relating to energy conservation.

[Approved by Governor June 3, 1976 Filed with
Secretary of State June 3, 1976]

LEGISLATIVE COUNSEL'S DIGEST

AB 2822, Nestande. Nuclear thermal electric sites.

Existing law, with prescribed exceptions, confers on the State Energy Resources Conservation and Development Commission the authority to certify thermal powerplants, electrical transmission lines, and their sites.

This bill would prohibit any nuclear fission thermal powerplant, including those exempted from certification, from being permitted land use in the state or, where applicable, from being certified by the commission until the commission finds that there has been developed and the United States through its authorized agency has approved and there exists a demonstrated technology or means for the disposal of high-level nuclear waste and 100 legislative days have passed since such findings of the commission have been filed with the Legislature and neither house of the Legislature has adopted a resolution disapproving such findings. In the event of passage of such resolution, the bill would delineate a procedure for reexamination of its findings by the commission, resubmission to the Legislature, and certification of nuclear powerplants if the Legislature does not act by statute to declare such findings null and void and take appropriate action within 100 legislative days of resubmission.

The bill would require the commission to continue to receive and process notices of intention and applications for certification, but would prohibit the commission from issuing a decision granting a certificate until the requirements of this bill have been met. "Technology or means for the disposal of high-level nuclear waste" would be defined by the bill.

It would specify that for purposes of the provisions of the bill the vested right to construct a nuclear thermal powerplant shall exist if, prior to the date on which the bill is chaptered, an electric utility has performed substantial construction on such powerplant and has incurred substantial expense for construction and for necessary materials for such powerplant and would include specific designated thermal powerplants and sites within such provision.

The bill would also specify that it would not become operative if Proposition 15 of the June 1976 election is adopted by the people.

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The people of the State of California do enact as follows:

SECTION 1. Section 25524.2 is added to the Public Resources Code, to read:

25524.2. No nuclear fission thermal powerplant, including any to which the provisions of this chapter do not otherwise apply, but excepting those exempted herein, shall be permitted land use in the state, or where applicable, be certified by the commission until both conditions (a) and (b) have been met:

(a) The commission finds that there has been developed and that the United States through its authorized agency has approved and there exists a demonstrated technology or means for the disposal of high-level nuclear waste.

(b) The commission has reported its findings and the reasons therefor pursuant to paragraph (a) to the Legislature. Such reports of findings shall be assigned to appropriate policy committees for review. The commission may proceed to certify nuclear fission thermal powerplants 100 legislative days after reporting its findings unless within those 100 legislative days either house of the Legislature adopts by a majority vote of its members a resolution disaffirming the findings of the commission made pursuant to paragraph (a).

A resolution of disaffirmance shall set forth the reasons for the action and shall provide to the extent possible, guidance to the commission as to an appropriate method of bringing the commission's findings into conformance with paragraph (a).

If a disaffirming resolution is adopted, the commission shall reexamine its original findings consistent with matters raised in the resolution. On conclusion of its reexamination, the commission shall reduce its findings to writing with the reasons therefor and shall transmit them to the Legislature.

If the findings are that the conditions of paragraph (a) have been met, the commission may proceed to certify nuclear fission thermal powerplants 100 legislative days after reporting its findings to the Legislature unless within those 100 legislative days both houses of the Legislature act by statute to declare the findings null and void and take appropriate action.

To allow sufficient time for the Legislature to act, the reports of findings of the commission shall be submitted to the Legislature at least six calendar months prior to the adjournment of the Legislature sine die.

(c) As used in this section, "technology or means for the disposal of high-level nuclear waste" means a method for the permanent and terminal disposition of high-level nuclear waste. It shall not necessarily require that facilities for the application of such technology and/or means be available at the time the commission makes its findings. Such disposition shall not necessarily preclude the possibility of an approved process for retrieval of such waste.

(d) The commission shall continue to receive and process notices of intention and applications for certification pursuant to this division.

but shall not issue a decision pursuant to Section 25523 granting a certificate until the requirements of this section have been met. All other permits, licenses, approvals or authorizations for the entry or use of the land, including orders of court, which may be required may be processed and granted by the governmental entity concerned but construction work to install permanent equipment or structures shall not commence until the requirements of this section have been met.

(e) Any nuclear fission powerplant is exempted from the provisions of this section if prior to the date on which this section is chaptered an electric utility has performed substantial construction on such powerplant and has incurred substantial expense for construction and for necessary materials for such powerplant, including, but not limited to, the following sites and facilities, with the associated estimated generating capacities:

(1) As designated in the report of the Pacific Gas and Electric Company submitted to the Public Utilities Commission on December 23, 1966, pursuant to Section 1001 of the Public Utilities Code, one nuclear thermal powerplant, having a generating capacity of 1,060 megawatts, commonly known as Diablo Canyon Unit 1, to be located in San Luis Obispo County.

(2) As designated in the report of the Pacific Gas and Electric Company submitted to the Public Utilities Commission on February 16, 1968, pursuant to Section 1001 of the Public Utilities Code, one nuclear thermal powerplant, having a generating capacity of 1,060 megawatts, commonly known as Diablo Canyon Unit 2, to be located in San Luis Obispo County.

(3) As designated in the report of the Southern California Edison Company and the San Diego Gas and Electric Company to the Public Utilities Commission on July 16, 1970, pursuant to Section 1001 of the Public Utilities Code, two nuclear thermal powerplants, having a generating capacity of 1,100 megawatts per unit, commonly known as San Onofre Unit 2 and San Onofre Unit 3, to be located in San Diego County.

SEC. 2. The provisions of this act shall not become operative if Proposition 15 of the June 1976 election is adopted by the people, whether or not this bill is chaptered before or after such adoption.

O